Nine Mile Point 3 Nuclear Power Plant

Combined License Application

Part 4: Technical Specifications and Bases

Revision 0
September, 2008

Introduction

The U.S. EPR generic Technical Specifications and Bases, provided in Chapter 16 of the U.S. EPR FSAR are used to develop the {NMP3NPP} Technical Specifications and Bases. Departures and supplements from the U.S. EPR generic Technical Specifications and Bases include replacing preliminary information provided in the generic Technical Specifications and Bases with {NMP3NPP} specific values and information, addressing and removing Reviewer's Notes, and use of a Setpoint Control Program.

As stated in Regulatory Guide 1.206, C.I.16, "Applicant-supplied information to fulfill COL information items for a certified design or, as discussed in Section C.IV.3.3.3 of this guide, to replace information bracketed in the generic TS and bases, is not considered a deviation from the generic TS and bases and does not require an exemption; however, the application should include the justification for such information." However, to incorporate the site-specific response, there were cases that required additional changes to non-bracketed text in the U.S. EPR generic Technical Specifications and Bases. While these changes satisfy the intent of the U.S. EPR generic Technical Specifications and Bases, these changes are addressed as departures.

These departures and supplements are addressed and justified in Section A of this part of the COL Application {and includes redline strikeout pages of the EPR generic Technical Specifications and Bases}. Section B provides a complete copy of the {NMP3NPP} Technical Specifications and Bases.

The U.S. EPR generic Technical Specifications and Bases have been submitted to the U.S. NRC for review and approval. While the U.S. EPR generic Technical Specifications and Bases are undergoing review, AREVA may issue changes to the U.S. EPR generic Technical Specifications and Bases for a number of reasons, e.g., to address comments or questions raised by the NRC or utilities using the U.S. EPR as their reactor design. These changes to the U.S. EPR generic Technical Specifications will be incorporated into the {NMP3NPP} Technical Specifications.

Section A – Departures and Supplements

1. Departure - Setpoint Control Program

    <u>Generic Technical Specifications (GTS) and Bases:</u>

    GTS Table 3.3.1-2 includes a bracketed Reviewer's Note that states "The values specified in brackets in the Limiting Trip Setpoint column are included for reviewer information only. A plant specific setpoint study will be conducted. The values of the Limiting Trip Setpoint will then be replaced after the completion of the study."

    <u>{NMP3NPP} Technical Specifications (TS) and Bases:</u>

    In order to address this bracketed Reviewer's Note, a Setpoint Control Program is adopted in the {NMP3NPP} TS. TS 5.5.18, Setpoint Control Program (SCP), is added to the TS. The TS requirements for the Setpoint Control Program establishes that Limiting Trip Setpoints (LTSPs), Nominal Trip Setpoints (NTSPs), Allowable Values (AVs), and As-Found Tolerance and As-Left Tolerance Bands for each of the required Technical Specification Instrument Functions in TS 3.3.1, "Protection Systems (PS)," shall be documented in the SCP. The TS requirements for the SCP also establish that the methods used to determine the Limiting Trip Setpoints (LTSPs), Nominal Trip Setpoints (NTSPs), Allowable Values (AVs), and As-Found Tolerance and As-Left Tolerance Bands for the required instrument functions shall be those included in NRC approved setpoint methodology documents. These NRC approved setpoint methodology documents are listed in TS 5.5.18. The TS requirements for the SCP also include the Technical Specification Task Force (TSTF)-493, "Clarify Application of Setpoint Methodology for LSSS Functions," guidance to provide assurance that the required instruments will always actuate safety functions at the point assumed in the applicable safety analyses. Finally, the TS for the SCP require the SCP to be provided, including any revisions or supplements, to the NRC on a periodic basis.

To implement the SCP, the following additional changes to GTS and Bases 3.3.1 are made in the {NMP3NPP} TS and Bases 3.3.1:

a. LTSPs specified in GTS Table 3.3.1-2 are replaced with a Setting Basis (i.e., the Analytical Limit or Design Limit, as defined in the Bases for each of the applicable instrument functions) in {NMP3NPP} Table 3.3.1-2. The LTSPs are relocated to the SCP (TS 5.5.18).

b. The reference to "Limiting Trip Setpoint" in GTS 3.3.1, Condition C, is replaced with a reference to "Setpoint Control Program requirements" in {NMP3NPP} TS 3.3.1, Condition C.

c. GTS SRs 3.3.1.4 and 3.3.1.6, which state "Perform CALIBRATION," are revised in {NMP3NPP} SRs 3.3.1.4 and 3.3.1.6 to "Perform CALIBRATION consistent with Specification 5.5.18, Setpoint Control Program (SCP)."

d. GTS Table 3.3.1-2 footnotes (b) and (c), which implement TSTF-493 guidance, are moved to {NMP3NPP} TS 5.5.18.c.1 and 5.5.18.c.2 and the subsequent footnotes in {NMP3NPP} Table 3.3.1-2 are relabeled.

e. The GTS Table 3.3.1-2 bracketed Reviewer's Note is deleted from {NMP3NPP} Table 3.3.1-2.

f. Corresponding changes to the GTS Bases 3.3.1, including deletion of the associated Reviewer's Notes, are made in the {NMP3NPP} Bases 3.3.1.

Justification:

Table 3.3.1-2 contains a Reviewer's Note which requires a plant specific setpoint study to be conducted and that the values of the Limiting Trip Setpoint be replaced after the completion of the study. However, the plant specific setpoint study cannot be completed until after selection of instrumentation. Nevertheless, instrumentation selection may not occur until after the approval of the COL application is granted. As an alternative approach, it is proposed that the Limiting Trip Setpoints be relocated to the Setpoint Control Program and that the Setting Basis (Analytical Limits or Design Limits, as applicable) for the required instrument functions be specified in the TS. The Setpoint Control Program is a TS required program and is consistent with the approach used for the TS required Core Operating Limits Report and the Pressure and Temperature Limits Report. In the case of the Core Operating Limits, the NRC approved relocation of cycle-specific parameter limits from the TS to the Core Operating Limits Report. The basis for acceptability of this approach was that the methodology for determining cycle-specific parameter limits is documented in NRC approved topical reports or in an NRC approved plant-specific submittal. As a consequence the NRC review of proposed changes to the TS for these cycle-specific parameter limits was primarily limited to confirmation that the updated limits were calculated using an NRC approved methodology and consistent with applicable limits of the safety analysis. The approach documented in the TS for the Core Operating Limits Report also allows the NRC to trend the parameter limit changes, if desired. The Core Operating Limits Report approach is documented in NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits for Technical Specifications," dated October 3, 1988, and is reflected in the current Improved Standard Technical Specifications (NUREG-1430 through NURG-1434). For the Setpoint Control Program, the TS require that that the Limiting Trip Setpoints be developed using NRC approved setpoint methodology. In addition, by specifying the Analytical Limits and Design Limits in the TS, assurance is provided that the Limiting Trip Setpoints are developed and maintained such that required instruments will always actuate safety functions at the point assumed in the applicable safety analyses. The approach documented in the TS for the Setpoint Control Program also allows the NRC to trend the parameter limit changes, if desired, since the TS requires the Setpoint Control Program to be submitted to the NRC prior to initial fuel load and periodically thereafter.

2. Departure - Error Correction – TS 3.1.1 Limiting Trip Setpoint Inequality Signs

<u>GTS:</u>

GTS 3.3.1, "Protection System (PS)," Table 3.3.1-2, includes Functions with Limiting Trip Setpoint values which are either missing inequality signs or have inequality signs specified incorrectly. These PS Functions are as follows.

Function A.3, High Neutron Flux Rate of Change (Power Change)

Function A.5, Low Saturation Margin

Function A.14, Steam Generator (SG) Pressure Drop

Function A.17, Low SG Level

Function A.18, High SG Level

Function A.19, High Containment Pressure

Function B.2.b, Main Feedwater Full Load Closure on High SG Level (Affected SGs)

Function B.2.c, Startup and Shutdown Feedwater Isolation on SG Pressure Drop (All SGs)

Function B.2.e, Startup and Shutdown Feedwater Isolation on High SG Level for Period of Time (Affected SGs)

Function B.8.a, Main Steam Isolation Valve (MSIV) Closure on SG Pressure Drop (All SGs)

Function B.9.a, Containment Isolation (Stage 1) on High Containment Pressure

Function B.9.c, Containment Isolation (Stage 2) on High-High Containment Pressure

Function B.9.d, Containment Isolation (Stage 1) on High Containment Radiation

{NMP3NPP} TS:

The Setting Basis values for the following Functions in {NMP3NPP} TS 3.3.1, "Protection System (PS)," Table 3.3.1-2, are revised to include the missing inequality signs and to correct the inequality signs, as required. Corresponding changes are made to the Bases, as required.

Function A.3, High Neutron Flux Rate of Change (Power Change)

Function A.5, Low Saturation Margin

Function A.14, Steam Generator (SG) Pressure Drop

Function A.17, Low SG Level

Function A.18, High SG Level

Function A.19, High Containment Pressure

Function B.2.b, Main Feedwater Full Load Closure on High SG Level (Affected SGs)

Function B.2.c, Startup and Shutdown Feedwater Isolation on SG Pressure Drop (All SGs)

Function B.2.e, Startup and Shutdown Feedwater Isolation on High SG Level for Period of Time (Affected SGs)

Function B.8.a, Main Steam Isolation Valve (MSIV) Closure on SG Pressure Drop (All SGs)

Function B.9.a, Containment Isolation (Stage 1) on High Containment Pressure

Function B.9.c, Containment Isolation (Stage 2) on High-High Containment Pressure

Function B.9.d, Containment Isolation (Stage 1) on High Containment Radiation

Justification:

The change corrects errors in the GTS to be consistent with the U.S. EPR design and analyses.

3. Departure - Error Correction – TS 3.3.1 Time Delays

GTS:

GTS 3.3.1, "Protection System (PS)," Table 3.3.1-2 includes Limiting Trip Setpoint values with time delays for Function A.18, High SG Level, and Function B.2.b, Main Feedwater Full Load Closure on High SG Level (Affected SGs).

{NMP3NPP} TS:

The time delays are removed from the {NMP3NPP} TS 3.3.1, "Protection System (PS)," Table 3.3.1-2 Setting Basis values for Function A.18, High SG Level, and Function B.2.b, Main Feedwater Full Load Closure on High SG Level (Affected SGs).

Justification:

The change corrects errors in the GTS to be consistent with the U.S. EPR design and analyses.

4. Departure - Error Correction – Location of Limiting Trip Setpoint/Setting Basis value in Core Operating Limits Report (COLR)

GTS:

GTS 3.3.1, "Protection System (PS)," Table 3.3.1-2 includes a Limiting Trip Setpoint value for Function B.11.b, CVCS Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating).

{NMP3NPP} TS:

The Setting Basis for {NMP3NPP} TS 3.3.1, "Protection System (PS)," Table 3.3.1-2 Function B.11.b, CVCS Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating), is revised to indicate that the value is "As specified in the COLR."

Justification:

The change corrects an error in the GTS. The values associated with the Limiting Trip Setpoint and Setting Basis are cycle-specific parameter values. As such, consistent with the Limiting Trip Setpoint specified in GTS 3.3.1, Table 3.3.1-2 for Function B.11.c, CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions, it is appropriate for the Limiting Trip Setpoint and Setting Basis for Function B.11.b to also be specified in the COLR.

5. {Departure – Incorporation of Site-Specific Information – Essential Service Water System

GTS and Bases:

GTS 3.7.8, "Essential Service Water (ESW) System," provides a set of requirements for a generic ESW System. GTS Bases 3.7.8 provides the supporting bases information.

NMP3NPP TS and Bases

NMP3NPP TS 3.7.8, Essential Service Water (ESW) System," and its associated Bases are revised to include: 1) Condition C and Required Action C.1 to address the situations when the ESW Emergency Makeup System (ESWEMS) Pond level or temperature is not within limit; 2) Re-number Condition C and Required Action C.1 of the GTS 3.7.8 as Condition D and Required Action D.1; 3) Add Surveillance Requirement 3.7.8.8 to require the periodic verification that the ESWEMS Pond average water temperature is within limit; and 4) Add Surveillance Requirement 3.7.8.9 to require the periodic verification that the ESWEMS Pond level is within limit.

Justification:

The additional Condition, Required Action, and Surveillance Requirements regarding the ESWEMS Pond water level and average water temperature are necessary to ensure that the ESWEMS remains OPERABLE.}

6. {Departure and Supplement - Incorporation of Site-Specific Information -Toxic Gas/Hazardous Chemical Protection

GTS and Bases:

GTS 3.7.10, "Control Room Emergency Filtration (CREF)," Required Action B.1, Required Action D.1, and associated Reviewer's Note, Bases 3.7.10 (Background, Applicable Safety Analyses, LCO, Actions sections), Bases 3.7.12, "Safeguard Building Controlled Area Ventilation System (SBVS)," Action B.1, and GTS 5.5.17, "Control Room Envelope Habitability Program," contain conceptual design information on toxic gas and hazardous chemicals.

NMP3NPP TS:

The information in NMP3NPP TS 3.7.10, "Control Room Emergency Filtration (CREF)," Required Action B.1, Required Action D.1, and associated Reviewer's Note, Bases 3.7.10 (Background, Applicable Safety Analyses, LCO, Actions and Surveillance Requirements section, sections), Bases 3.7.12, "Safeguard Building Controlled Area Ventilation System (SBVS)," Action B.1, and TS 5.5.17, "Control Room Envelope Habitability Program," pertaining to toxic gas and hazardous chemicals is removed.

Justification:

The conceptual design information and Reviewer's Note associated with toxic gas and hazardous chemical protection are not applicable in NMP3NPP. Toxic gas and hazardous chemical protection for the CREF is not required for NMP3NPP based on the site-specific evaluation provided in NMP3NPP FSAR Sections 2.2.3 and 6.4.4.}

7. Departure and Supplement - Incorporation of Site-Specific Information -Spent Fuel Storage Pool Boron Concentration

GTS:

GTS and Bases 3.7.15, "Spent Fuel Storage Pool Boron Concentration," includes a Reviewer's Note that states "The design of the spent fuel storage racks is to be provided by the COLA applicant. The required boron concentration will be provided as part of the spent fuel rack design."

{NMP3NPP} TS:

{NMP3NPP} TS and Bases 3.7.15, "Spent Fuel Storage Pool Boron Concentration," are revised to remove the Reviewer's Note and include the {NMP3NPP} specific spent fuel storage pool boron concentration in the TS and Bases. The associated Bases (all sections) are also revised to include a discussion of the bases for the specific spent fuel storage pool boron concentration requirements.

Justification:

The {NMP3NPP} specific TS and Bases 3.7.15 are revised to reflect the results of the analyses of the {NMP3NPP} specific spent fuel rack design described in the Holtec Topical Report for the design and analyses of the U.S. EPR spent fuel storage racks.

8. Departure and Supplement - Incorporation of Site-Specific Information -Spent Fuel Storage

GTS:

GTS and Bases 3.7.16, "Spent Fuel Storage," includes a Reviewer's Note that states "The design of the spent fuel storage racks is to be provided by the COLA applicant. The required spent fuel storage configuration will be provided as part of the spent fuel rack design."

{NMP3NPP} TS:

{NMP3NPP} TS and Bases 3.7.16, "Spent Fuel Storage," are revised to remove the Reviewer's Note and include the {NMP3NPP} specific spent fuel storage requirements in the TS and Bases, including the addition of Figure 3.7.16-1, "Fuel Assembly Burnup Requirements for Region 2." The associated Bases (all sections) are also revised to include a discussion of the bases for the specific spent fuel storage requirements.

Justification:

The {NMP3NPP} specific TS and Bases 3.7.16 are revised to reflect the results of the design and analyses of the {NMP3NPP} specific spent fuel racks described in the Holtec Topical Report for the design and analyses of the U.S. EPR spent fuel storage racks.

9. Supplement - Site Location

GTS:

GTS 4.1, "Site Location," contains a bracketed requirement for the COL application to provide site specific information for Section 4.1, "Site Location."

{NMP3NPP} TS:

The bracketed information in {NMP3NPP} TS 4.1, Site Location," is removed and replace the information with {NMP3NPP} specific information regarding the site location.

Justification:

The site location information provided is consistent with the {NMP3NPP} FSAR description of site location.

10. Supplement - Incorporation of Site-Specific Information - Fuel Storage Rack Uncertainties

GTS:

GTS 4.3.1, "Criticality," includes a Reviewer's Note that states "Storage rack uncertainties are discussed in the FSAR of COLA Section 9.1." In addition, GTS 4.3.1.1.b, 4.3.1.2.b, and 4.3.1.2.c include brackets around the reference (i.e., FSAR) for the location of the description of the spent and new fuel storage rack uncertainties.

{NMP3NPP} TS:

The Reviewers Note in {NMP3NPP} TS 4.3.1, Criticality," is removed and the brackets are removed from around the GTS reference for the location of the description of the spent and new fuel storage rack uncertainties (i.e., FSAR).

Justification:

{NMP3NPP} FSAR Section 9.1 will include, by incorporation of reference, the description of spent and new fuel storage rack uncertainties.

11. Departure and Supplement – Incorporation of Site-Specific Information - Spent Fuel Storage Dimensions and Restrictions

GTS:

GTS 4.3.1, "Criticality," Section 4.3.1.1 contains bracketed requirements for the COL application to provide the spent fuel rack center to center distance between fuel assemblies.

{NMP3NPP} TS:

{NMP3NPP} TS 4.3.1.1.c is revised to include the spent fuel rack center to center distances between fuel assemblies stored in Region 1 and for fuel assemblies stored in Region 2 of the {NMP3NPP} spent fuel racks. In addition, restrictions on the storage of fuel assemblies are established in TS 4.3.1.1.d and 4.3.1.1.e. Figure 4.3-1, "Discrete Two Region Spent Fuel Pool Rack Layout" is added to identify the storage cells associated with Region 1 and the storage cells associated with Region 2.

Justification:

{NMP3NPP} specific TS 4.3.1.1 requirements are revised to reflect the results of the design and analyses of the {NMP3NPP} specific spent fuel racks described in the Holtec Topical Report for the design and analyses of the U.S. EPR spent fuel storage racks.

12. Departure and Supplement – Incorporation of Site-Specific Information - New Fuel Storage Dimensions

    GTS:

    GTS 4.3.1, "Criticality," Section 4.3.1.2 contains bracketed requirements for the COL application to provide the new fuel rack center to center distance between fuel assemblies.

    {NMP3NPP} TS:

    {NMP3NPP} TS 4.3.1.2.d is revised to include the new fuel rack center to center distances between fuel assemblies stored in the {NMP3NPP} new fuel racks.

    Justification:

    {NMP3NPP} specific TS 4.3.1.2 requirements are revised to reflect the results of the design and analyses of the {NMP3NPP} specific new fuel racks described in the Holtec Topical Report for the design and analyses of the U.S. EPR new fuel storage racks.

13. Supplement – Incorporation of Site-Specific Information - Spent Fuel Storage Capacity

    GTS:

    GTS 4.3.3, "Capacity," contains bracketed requirements for the COL application to provide the capacity for spent fuel storage in the spent fuel storage pool.

    {NMP3NPP} TS:

    {NMP3NPP} TS 4.3.3 is revised to include capacity for spent fuel storage in the {NMP3NPP} spent fuel storage pool.

    Justification:

    {NMP3NPP} specific TS 4.3.3 requirements are revised to reflect the spent fuel storage capacity resulting from the design and analyses of the {NMP3NPP} specific spent fuel racks described in the Holtec Topical Report for the design and analyses of the U.S. EPR spent fuel storage racks.

14. Departure and Supplement – Incorporation of Site-Specific Information - Generic Organizational Titles

    GTS:

    GTS 5.1, "Responsibility," includes two Reviewer's Notes related titles for members of the unit staff.

    {NMP3NPP} TS:

    {NMP3NPP} TS 5.1 is revised to remove the Reviewer's Notes and replace them with a note requiring that the organizational positions listed in the Administrative Controls section have corresponding plant-specific titles specified in the Final Safety Analysis Report (FSAR).

Justification:

The use of generic titles in the TS, and the inclusion of plant-specific, corresponding titles in the FSAR, is consistent with Improved Standard Technical Specifications, Revision 3.1 of NUREG-1430 through NUREG-1434.

15. Supplement – Incorporation of Site-Specific Information - Non-licensed Operators for Two Units

GTS:

GTS 5.2.2, "Unit Staff," contains a Reviewer's Note specifying the number of non-licensed operators required for two units when both units are shutdown or defueled.

{NMP3NPP} TS:

{NMP3NPP} TS 5.2.2, "Unit Staff," is revised to remove the Reviewer's Note.

Justification:

The {NMP3NPP} U.S. EPR is a single unit facility.

16. Supplement – Incorporation of Site-Specific Information - Minimum Qualifications of Unit Staff

GTS:

GTS 5.3, "Unit Staff Qualifications," contains a Reviewer's Note on the specification of the minimum qualifications of the unit staff.

{NMP3NPP} TS:

{NMP3NPP} TS 5.3, "Unit Staff Qualifications," is revised to remove the Reviewer's Note.

Justification:

The unit staff qualifications standards are provided consistent with the {NMP3NPP} FSAR, including FSAR Section 13.2, for the stated exception regarding cold license operator candidates.

17. Supplement – Incorporation of Site-Specific Information Temporary Outdoor Liquid Radwaste Storage Tanks

GTS:

GTS 5.5.11, "Gaseous Waste Processing System Radioactivity Monitoring Program," contains a Reviewer's Note for applicants incorporating outdoor liquid radioactive waste storage tanks in their design.

{NMP3NPP} TS:

{NMP3NPP} TS 5.11, “Gaseous Waste Processing System Radioactivity Monitoring Program,” is revised to remove the Reviewer’s Note.

Justification:

The {NMP3NPP} specific design does not include outdoor liquid radioactive waste storage tanks.

18. Supplement - Incorporation of Site-Specific Information -Containment Bypass Leakage Paths

GTS:

GTS 5.5.15, “Containment Leakage Rate Testing Program,” contains a Reviewer’s Note indicating that, as discussed in FSAR Section 6.2.6, the U.S. EPR has no penetrations that are classified as bypass leakage paths.

{NMP3NPP} TS:

{NMP3NPP} TS 5.5.15, “Containment Leakage Rate Testing Program,” is revised to remove the Reviewer’s Note.

Justification:

The {NMP3NPP} specific design has no penetrations that are classified as bypass leakage paths. This design information is reflected in FSAR Section 6.2.6 and does not need to be repeated in the TS.

19. {Supplement - Incorporation of Site-Specific Information -Multi-Unit Site Reporting Options

GTS:

GTS 5.6.1, “Annual Radiological Environmental Operating Report,” and GTS 5.6.2, “Radioactive Effluent Release Report” contain Reviewer’s Notes to allow a single report submittal for all units at a multi-unit site.

NMP3NPP TS:

NMP3NPP TS 5.6.1, “Annual Radiological Environmental Operating Report,” and GTS 5.6.2, “Radioactive Effluent Release Report” are revised to remove the Reviewer’s Notes.

Justification:

The allowance for submittal of a single report for both units at the Callaway Plant site is not being pursued at this time.}

20. Supplement – Incorporation of Site-Specific Information - Application of Topical Reports for Surveillance Frequency Extensions

GTS:

GTS Bases 3.3.1, "Protection System (PS)," in the Surveillance Requirements section, includes a Reviewer's Note that states "In order for a plant to take credit for topical reports as the basis for justifying Frequencies, topical reports must be supported by an NRC staff SER that establishes the acceptability of each topical report for that unit."

{NMP3NPP} TS:

{NMP3NPP} Bases 3.3.1, "Protection System (PS)," in the Surveillance Requirements section, is revised to remove the Reviewer's Note.

Justification:

The specified Frequencies in the {NMP3NPP} TS 3.3.1 are based on the Frequencies specified in GTS 3.3.1. {NMP3NPP} TS 3.3.1 does not credit topical reports as the basis for justifying Surveillance Frequencies.

21. Supplement – Incorporation of Site-Specific Information - Containment Leakage Acceptance Criteria

GTS Bases:

GTS Bases 3.6.1, "Containment," contains a Reviewer's Note, in the Bases for SR 3.6.1.1 indicating that Regulatory Guide 1.163 and NEI 94-01 contain acceptance criteria for containment leakage which may be reflected in the Bases.

{NMP3NPP} TS Bases:

{NMP3NPP} TS Bases 3.6.1, "Containment," is revised to remove the Reviewer's Note.

Justification:

The {NMP3NPP} Containment Leakage Rate Testing Program is conducted as required by TS 5.5.15, "Containment Leakage Rate Testing Program," and U.S. EPR FSAR 6.2.6, "Containment Leakage Testing." U.S. EPR FSAR 6.2.6 was developed to be consistent with Regulatory Guide 1.163 and NEI 94-01. Therefore, the information reflected in the Reviewer's Note does not need to be included in the Bases.

22. Departure and Supplement - Incorporation of Site-Specific Information - Seismic Category 1 Essential Service Water Emergency Makeup System

GTS Bases:

GTS Bases 3.7.8, "Essential Service Water (ESW) System," contains a bracketed requirement in the Background section for the COL application to provide site specific information for the seismic Category 1 to the ESW System.

{NMP3NPP} TS Bases:

{NMP3NPP} Bases 3.7.8, “Essential Service Water (ESW) System,” is revised, in the Background section, to remove the bracketed requirement and replace the information with {NMP3NPP} specific information regarding the seismic Category 1 ESW System makeup. In addition, other sections of the {NMP3NPP} Bases are revised to incorporate the site-specific information.

Justification:

The site-specific information provided is consistent with the {NMP3NPP} FSAR Section 9.2 description of seismic Category 1 ESW System makeup.

23. Supplement – Incorporation of Site-Specific Information - Definition of OPERABLE Essential Service Water Emergency Makeup System

GTS Bases:

GTS Bases 3.7.8, “Essential Service Water (ESW) System,” contains a bracketed requirement in the LCO section for the COL application to provide site specific information for the definition of an OPERABLE ESW System makeup source.

{NMP3NPP} TS Bases:

{NMP3NPP} Bases 3.7.8, “Essential Service Water (ESW) System,” is revised, in the LCO section, to remove the bracketed requirement and replace the information with {NMP3NPP} specific information regarding the definition of an OPERABLE ESW System makeup source.

Justification:

The site specific information provided is consistent with the {NMP3NPP} FSAR Section 9.2 description of seismic Category 1 ESW System makeup source.

24. Departure and Supplement - Safeguard Building or Fuel Building Boundary Inoperability

GTS Bases:

GTS Bases 3.7.12, “Safeguard Building Controlled Area Ventilation System (SBVS),” contains a Reviewer’s Note that, in the Actions section for Required Action B.1, indicates that the adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of intentional or unintentional entry into Condition B.

{NMP3NPP} TS Bases:

{NMP3NPP} Bases 3.7.12, “Safeguard Building Controlled Area Ventilation System (SBVS),” is revised, in the Actions section for Required Action B.1, to remove the Reviewer’s Note and modify the Bases discussion for Required Action B.1 to include the required commitment.

Justification:

The site specific commitment provided is consistent with the requirements in the Reviewer's Note for adoption of the allowance provided in Condition B of TS 3.7.12, "Safeguard Building Controlled Area Ventilation System (SBVS)."

{The following pages are markups of the EPR generic Technical Specifications and Bases identifying the above departures and supplements.}

3.3 INSTRUMENTATION

3.3.1 Protection System (PS)

LCO 3.3.1 The PS sensors, manual actuation switches, signal processors, and actuation devices specified in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

-- NOTE --
Separate Condition entry is allowed for each sensor, manual actuation switch, signal processor, and actuation device.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more sensors inoperable.	A.1 --------------NOTE--------------- Only applicable for Table 3.3.1-1, Component A.21. -------------------------------------	
	Place inoperable sensor in trip.	1 hour
	<u>AND</u>	
	A.2 --------------NOTE--------------- Not applicable for Table 3.3.1-1, Component A.21. -------------------------------------	
	Place inoperable sensor in lockout.	4 hours
B. One or more manual actuation switches inoperable.	B.1 Restore manual actuation switch to OPERABLE status.	48 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more acquisition and processing units (APUs) inoperable due to the ~~Limiting Trip Setpoint (LTSP)~~Setpoint Control Program requirements for one or more Trip/Actuation Functions not met.	C.1 --------------NOTE--------------- Only applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
	Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown," for emergency diesel generator (EDG) made inoperable by inoperable APU.	1 hour
	AND	
	C.2 --------------NOTE--------------- Not applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
	Place the Trip/Actuation Function in the associated APU in lockout.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more signal processors inoperable for reasons other than Condition C.	D.1	--------------NOTE--------------- Only applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
		Enter applicable Conditions and Required Actions of LCO 3.8.1 and LCO 3.8.2 for EDG made inoperable by inoperable APU.	1 hour
	AND		
	D.2	--------------NOTE--------------- Not applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
		Place inoperable signal processor in lockout.	4 hours
E. One or more actuation devices inoperable.	E.1	Restore actuation device to OPERABLE status.	48 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR Minimum functional capability specified in Table 3.3.1-1 not maintained.	F.1	Enter the Condition referenced in Table 3.3.1-1.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. As required by Required Action F.1 and referenced in Table 3.3.1-1.	G.1	Reduce THERMAL POWER to < 70% RTP.	2 hours
H. As required by Required Action F.1 and referenced in Table 3.3.1-1.	H.1	Reduce THERMAL POWER to < 10% RTP.	6 hours
I. As required by Required Action F.1 and referenced in Table 3.3.1-1.	I.1	Be in MODE 2.	6 hours
J. As required by Required Action F.1 and referenced in Table 3.3.1-1.	J.1	Be in MODE 3.	6 hours
K. As required by Required Action F.1 and referenced in Table 3.3.1-1.	K.1	Be in MODE 3.	6 hours
	AND		
	K.2	Open the reactor trip breakers.	6 hours
L. As required by Required Action F.1 and referenced in Table 3.3.1-1.	L.1	Be in MODE 3.	6 hours
	AND		
	L.2	Reduce pressurizer pressure to < 2005 psia.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
M. As required by Required Action F.1 and referenced in Table 3.3.1-1.	M.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	M.2	Be in MODE 4.	12 hours
N. As required by Required Action F.1 and referenced in Table 3.3.1-1.	N.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	N.2	Be in MODE 5.	36 hours
O. As required by Required Action F.1 and referenced in Table 3.3.1-1.	O.1	Declare associated EDG inoperable.	Immediately
P. As required by Required Action F.1 and referenced in Table 3.3.1-1.	P.1	Declare associated Chemical and Volume Control System isolation valve(s) inoperable.	Immediately
Q. As required by Required Action F.1 and referenced in Table 3.3.1-1.	Q.1	Declare associated Pressurizer Safety Relief Valve(s) inoperable.	Immediately
R. As required by Required Action F.1 and referenced in Table 3.3.1-1.	R.1	Declare both Control Room Emergency Filtration trains inoperable.	Immediately
S. As required by Required Action F.1 and referenced in Table 3.3.1-1.	S.1	Open reactor trip breakers.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
T. As required by Required Action F.1 and referenced in Table 3.3.1-1.	T.1 Declare associated Actuation Logic Units inoperable.	Immediately
	AND	
	T.2 Open reactor trip breakers.	1 hour

SURVEILLANCE REQUIREMENTS

---NOTES---

1. Refer to Table 3.3.1-1 to determine which SRs apply for each sensor, manual actuation switch, signal processor, or actuation device.

2. When a sensor, manual actuation switch, signal processor, or actuation device is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Trip/Actuation Function maintains functional capability.

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 20% RTP. --	
	Compare results of calorimetric heat balance calculation to power range division output. Adjust power range division output if calorimetric heat balance calculations results exceed power range division output by more than +2% RTP.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	-----NOTE----- Not required to be performed until 12 hours after THERMAL POWER ≥ 20% RTP. ----- Perform CALIBRATION.	15 effective full power days
SR 3.3.1.3	Perform ACTUATION DEVICE OPERATIONAL TEST.	31 days
SR 3.3.1.4	Perform CALIBRATION consistent with Specification 5.5.18, “Setpoint Control Program (SCP).”	92 days
SR 3.3.1.5	Perform a SENSOR OPERATIONAL TEST.	24 months
SR 3.3.1.6	-----NOTE----- Neutron detectors are excluded from CALIBRATION. ----- Perform a CALIBRATION~~.~~ consistent with Specification 5.5.18, “Setpoint Control Program (SCP).”	24 months
SR 3.3.1.7	Perform EXTENDED SELF TESTS.	24 months
SR 3.3.1.8	Perform ACTUATION DEVICE OPERATIONAL TEST.	24 months
SR 3.3.1.9	Verify setpoints properly loaded in APUs.	24 months

Table 3.3.1-1 (page 1 of 3)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUMUM REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
A. Sensors					
1. 6.9 kV Bus Voltage	3 per EDG	1,2,3,4,(a)	2 per EDG	O	SR 3.3.1.5 SR 3.3.1.6
2. Boron Concentration - Chemical and Volume Control System (CVCS) Charging Line	4	3[(b)],4[(b)],5,6	2	P	SR 3.3.1.4 SR 3.3.1.5
3. Boron Temperature - CVCS Charging Line	4	3[(b)],4[(b)],5,6	2	P	SR 3.3.1.5 SR 3.3.1.6
4. CVCS Charging Line Flow	4	3[(b)],4[(b)],5[(b)]	2	P	SR 3.3.1.5 SR 3.3.1.6
5. Cold Leg Temperature (Narrow Range)	4	≥ 10% RTP	3	H	SR 3.3.1.5 SR 3.3.1.6
6. Cold Leg Temperature (Wide Range)	4	1,2[(c)]	3	J	SR 3.3.1.5 SR 3.3.1.6
	4	3,4,5,6[(b)]	2	P	SR 3.3.1.5 SR 3.3.1.6
7. Containment Pressure	4 per area	1,2,3	3 per area	M	SR 3.3.1.5 SR 3.3.1.6
8. Hot Leg Pressure (Wide Range)	4	1,2,3	3	M	SR 3.3.1.5 SR 3.3.1.6
	4	(d)	2	Q	SR 3.3.1.5 SR 3.3.1.6
9. Hot Leg Temperature (Narrow Range)	4 per division, 4 divisions	1,2[(c)]	3 per division, 3 divisions	J	SR 3.3.1.5 SR 3.3.1.6
10. Hot Leg Temperature (Wide Range)	4	3[(e)]	3	M	SR 3.3.1.5 SR 3.3.1.6

(a) When associated EDG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

(b) With three or more reactor coolant pumps (RCPs) in operation.

(c) ≥ 10^{-5}% power on the intermediate range detectors.

(d) When Pressurizer Safety Relief Valves (PSRVs) are required to be OPERABLE per LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

(e) When Table 3.3.1-2, Trip/Actuation Function B.3.a is disabled.

Table 3.3.1-1 (page 2 of 3)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUMUM REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
11. Intermediate Range	4	1(f),2,3(g)	3	K	SR 3.3.1.5 SR 3.3.1.6
12. Power Range	2 per division, 4 divisions	1,2,3(g)	2 per division, 3 divisions	K	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6
13. Pressurizer Level (Narrow Range)	4	1,2,3	3	M	SR 3.3.1.5 SR 3.3.1.6
14. Pressurizer Pressure (Narrow Range)	4	1,2,3(h)	3	L	SR 3.3.1.5 SR 3.3.1.6
15. Radiation Monitor - Containment High Range	4	1,2,3,4	3	K	SR 3.3.1.5 SR 3.3.1.6
16. Radiation Monitor - Control Room HVAC Intake Activity	4	1,2,3,4	3	N	SR 3.3.1.5 SR 3.3.1.6
	4	5,6,(i)	3	R	SR 3.3.1.5 SR 3.3.1.6
17. RCP Current	3 per RCP	1,2,3	2 per RCP	M	SR 3.3.1.5 SR 3.3.1.6
18. RCP Delta P Sensors	2 per RCP	1,2,3	1 per RCP	M	SR 3.3.1.5 SR 3.3.1.6
19. RCP Speed	4	≥ 10% RTP	3	H	SR 3.3.1.5 SR 3.3.1.6
20. Reactor Coolant System (RCS) Loop Flow	4 per loop	1,2(c)	3 per loop	J	SR 3.3.1.5 SR 3.3.1.6
21. Reactor Trip Circuit Breaker Position Indication	4	1,2(g),3g)	3	M	SR 3.3.1.5 SR 3.3.1.8
22. Self-Powered Neutron Detectors	72	≥ 10% RTP	51	H	SR 3.3.1.2 SR 3.3.1.5

(c) ≥ 10^{-5} % power on the intermediate range detectors.

(f) ≤ 10% RTP.

(g) With the Reactor Control, Surveillance and Limitation (RCSL) System capable of withdrawing a Rod Cluster Control Assembly (RCCA) or one or more RCCAs not fully inserted.

(h) With pressurizer pressure ≥ 2005 psia.

(i) During movement of irradiated fuel assemblies.

Table 3.3.1-1 (page 3 of 3)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUMUM REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
23. Steam Generator (SG) Level (Narrow Range)	4 per SG	1,2(j),3(j)	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6
24. SG Level (Wide Range)	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6
25. SG Pressure	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6
B. Manual Actuation Switches					
1. Reactor Trip	4	1,2,3(g)	3	K	SR 3.3.1.8
	4	4(g),5(g)	3	S	SR 3.3.1.8
2. Safety Injection System (SIS) Actuation	4	1,2,3,4	3	N	SR 3.3.1.8
3. SG Isolation	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.8
C. Signal Processors					
1. Remote Acquisition Units (RAUs)	2 per division, 4 divisions	≥ 10% RTP	1 per division, 4 divisions	H	SR 3.3.1.5 SR 3.3.1.7
2. Acquisition and Processing Units (APUs)	5 per division, 4 divisions	Refer to Table 3.3.1-2	Refer to Table 3.3.1-2	Refer to Table 3.3.1-2	SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9
3. Actuation Logic Units (ALUs)	4 per division, 4 divisions	1,2,3,4	3 per division, 4 divisions	N	SR 3.3.1.5 SR 3.3.1.7
	4 per division, 4 divisions	5,6,(i)	3 per division, 4 divisions	T	SR 3.3.1.5 SR 3.3.1.7
D. Actuation Devices					
1. Reactor Coolant Pump Bus and Trip Breakers	2 per pump	1,2,3,4	1 per pump	N	SR 3.3.1.8
2. Reactor Trip Circuit Breakers	4	1,2,3(g)	3	K	SR 3.3.1.3
3. Reactor Trip Contactors	4 per set, 23 sets	1,2,3(g)	3 per set, 23 sets	K	SR 3.3.1.3

(g) With the RCSL capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(i) During movement of irradiated fuel assemblies.

(j) Except when all main feedwater (MFW) isolation valves are closed.

Table 3.3.1-2 (page 1 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[(a)]	SETTING BASIS~~LIMITING TRIP SETPOINT[(b)(c)]~~	CONDITION
A.	Reactor Trip				
1.a.	Low Departure from Nucleate Boiling Ratio (DNBR)	≥ 10% RTP	3 divisions	(b~~d~~)	H
1.b.	Low DNBR and Imbalance or Rod Drop	≥ 10% RTP	3 divisions	(b~~d~~)	H
1.c.	Variable Low DNBR and Rod Drop	≥ 10% RTP	3 divisions	(b~~d~~)	H
1.d.	Low DNBR - High Quality	≥ 10% RTP	3 divisions	(b~~d~~)	H
1.e.	Low DNBR - High Quality and Imbalance or Rod Drop	≥ 10% RTP	3 divisions	(b~~d~~)	H
2.	High Linear Power Density	≥ 10% RTP	3 divisions	(b~~d~~)	H
3.	High Neutron Flux Rate of Change (Power Range)	1,2,3[(c~~e~~)]	3 divisions	≥ 13~~1~~% RTP	K
4.	High Core Power Level	1,2[(d~~f~~)]	3 divisions	≤ 116.7~~05~~% RTP	J
5.	Low Saturation Margin	1,2[(d~~f~~)]	3 divisions	≥ 0~~[430]~~ Btu/lb	J
6.a.	Low-Low Reactor Coolant System (RCS) Loop Flow Rate in One Loop	≥ 70% RTP	3 divisions	≥ 50~~4~~% Nominal Flow	G
6.b.	Low RCS Loop Flow Rate in Two Loops	≥ 10% RTP	3 divisions	≥ 86~~90~~% Nominal Flow	H
7.	Low Reactor Coolant Pump (RCP) Speed	≥ 10% RTP	3 divisions	≥ 92~~3~~% Nominal Speed	H
8.	High Neutron Flux (Intermediate Range)	1[(e~~g~~)],2,3[(c~~e~~)]	3 divisions	≤ 2~~1~~5% RTP	K

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

~~(b) If the as-found setpoint is outside its predefined as-found tolerance, then the Trip/Actuation Function shall be evaluated to verify that it is functioning as required before returning the Trip/Actuation Function to service.~~

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(b~~d~~) As specified in the COLR.

(c~~e~~) With the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(d~~f~~) ≥ 10^{-5}% power on the intermediate range detectors.

(e~~g~~) ≤ 10% RTP.

Table 3.3.1-2 (page 2 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[(a)]	SETTING BASIS~~LIMITING TRIP SETPOINT~~~~[(b)(c)]~~	CONDITION
9.	Low Doubling Time (Intermediate Range)	1[(e~~g~~)],2,3[(c~~e~~)]	3 divisions	≥ 1~~2~~0 Sec.	K
10.	Low Pressurizer Pressure	≥ 10% RTP	3 divisions	≥ 1950~~2005~~ psia	H
11.	High Pressurizer Pressure	1,2	3 divisions	≤ 247~~015~~ psia	J
12.	High Pressurizer Level	1,2	3 divisions	≤ 83~~75~~% Measuring Range	J
13.	Low Hot Leg Pressure	1,2,3[(c~~e~~)(f~~h~~)]	3 divisions	≥ 189~~200~~5 psia	L
14.	Steam Generator (SG) Pressure Drop	1,2	3 divisions	≥ 29 psi/min; 177~~02~~ psi<ss; Max 1088 psia	J
15.	Low SG Pressure	1,2,3[(c~~e~~)(f~~h~~)]	3 divisions	≥ 650~~725~~ psia	M
16.	High SG Pressure	1	3 divisions	≤ 1460~~385~~ psia	I
17.	Low SG Level	1,2	3 divisions	≥~~≤~~ 3.5~~20~~% Narrow Range	J
18.	High SG Level	1,2	3 divisions	≤~~≥~~ 80.5~~69~~% Narrow Range ~~for 10 sec.~~	J
19.	High Containment Pressure	1,2	3 divisions	≤ 1~~8.7~~9.2 psia	J

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

~~(b) If the as-found setpoint is outside its predefined as-found tolerance, then the Trip/Actuation Function shall be evaluated to verify that it is functioning as required before returning the Trip/Actuation Function to service.~~

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(c~~e~~) With the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(e~~g~~) ≤ 10% RTP.

(f~~h~~) With pressurizer pressure ≥ 2005 psia.

Table 3.3.1-2 (page 3 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY(a)	SETTING BASIS~~LIMITING TRIP SETPOINT(b)(c)~~	CONDITION
B.	ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) SIGNALS				
1.	Turbine Trip on Reactor Trip (RT)	1	3 divisions	RT for 1 sec.	I
2.a.	Main Feedwater Full Load Closure on Reactor Trip (All SGs)	1,2(g~~i~~)	3 divisions	NA	J
2.b.	Main Feedwater Full Load Closure on High SG Level (Affected SGs)	1,2(g~~i~~),3(g~~i~~)	3 divisions	≤~~≥~~ 80.5~~69~~% Narrow Range ~~for 10 sec.~~	M
2.c.	Startup and Shutdown Feedwater Isolation on SG Pressure Drop (All SGs)	1,2(h~~j~~),3(h~~j~~)	3 divisions	≥ 29 psi/min; 322~~247~~ psi<ss; Max 943 psia	M
2.d.	Startup and Shutdown Feedwater Isolation on Low SG Pressure (All SGs)	1,2(h~~j~~),3(f~~h~~)(h~~j~~)	3 divisions	≥ 505~~80~~ psia	L
2.e.	Startup and Shutdown Feedwater Isolation on High SG Level for Period of Time (Affected SGs)	1,2(h~~j~~),3(h~~j~~)	3 divisions	≤~~≥~~ 66.5~~69~~% Narrow Range for 10 sec.	M
3.a.	Safety Injection System (SIS) Actuation on Low Pressurizer Pressure	1,2,3(j~~h~~)	3 divisions	≥ ~~1668~~ 1613 psia	L
3.b.	SIS Actuation on Low Delta Psat	3(i~~k~~)	3 divisions	≥ ~~220~~ 39 psia	M
4.	RCP Trip on Low Delta P across RCP with SIS Actuation	1,2,3	3 divisions	≥ ~~80~~75% Nominal Pressure	M
5.	Partial Cooldown Actuation on SIS Actuation	1,2,3	3 divisions	NA	M

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

~~(b) If the as-found setpoint is outside its predefined as-found tolerance, then the Trip/Actuation Function shall be evaluated to verify that it is functioning as required before returning the Trip/Actuation Function to service.~~

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(f~~h~~) With pressurizer pressure ≥ 2005 psia.

(g~~i~~) Except when all MFW full load isolation valves are closed.

(h~~j~~) Except when all MFW low load isolation valves are closed.

(i~~k~~) When Trip/Actuation Function B.3.a is disabled.

Table 3.3.1-2 (page 4 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[a]	SETTING BASIS~~LIMITING TRIP SETPOINT[b][c]~~	CONDITION
6.a.	Emergency Feedwater System (EFWS) Actuation on Low-Low SG Level (All SGs)	1,2,3	3 divisions	≥ ~~40~~29% Wide Range	M
6.b.	EFWS Actuation on Loss of Offsite Power (LOOP) and SIS Actuation (All SGs)	1,2	3 divisions	NA	J
6.c.	EFWS Isolation on High SG Level (Affected SG)	1,2,3	3 divisions	≤ ~~89~~98% Wide Range	M
7.a.	Main Steam Relief Train (MSRT) Actuation on High SG Pressure	1,2,3	3 divisions	≤ 1460~~1385~~ psia	M
7.b.	MSRT Isolation on Low SG Pressure	1,2,3[f~~h~~]	3 divisions	≥ 505~~80~~ psia	L
8.a.	Main Steam Isolation Valve (MSIV) Closure on SG Pressure Drop (All SGs)	1,2,3	3 divisions	≥ 29 psi/min; ~~102~~ 177 psi<ss; Max 1088 psia	M
8.b.	MSIV Closure on Low SG Pressure (All SGs)	1,2,3[j~~l~~]	3 divisions	≥ 650~~725~~ psia	L
9.a.	Containment Isolation (Stage 1) on High Containment Pressure	1,2,3	3 divisions	≤ 1~~8.7~~9.2 psia	M
9.b.	Containment Isolation (Stage 1) on SIS Actuation	1,2,3,4	3 divisions	NA	N
9.c.	Containment Isolation (Stage 2) on High-High Containment Pressure	1,2,3	3 divisions	≤ ~~[~~38~~6~~.3~~]~~ psia	M
9.d.	Containment Isolation (Stage 1) on High Containment Radiation	1,2,3,4	3 divisions	≤ 100 x background	N

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

~~(b) If the as-found setpoint is outside its predefined as-found tolerance, then the Trip/Actuation Function shall be evaluated to verify that it is functioning as required before returning the Trip/Actuation Function to service.~~

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(f~~h~~) With pressurizer pressure ≥ 2005 psia.

(j~~l~~) Except when all MSIVs are closed.

Table 3.3.1-2 (page 5 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[a]	SETTING BASIS~~LIMITING TRIP SETPOINT[(b)(c)]~~	CONDITION
10.a. Emergency Diesel Generator (EDG) Start on Degraded Grid Voltage	1,2,3,4,(k~~m~~)	NA	≥ ~~6210~~ 6089 V and ≤ ~~6350~~ 6486 V; ≥ ~~7~~ 6.5 sec. and ≤ ~~11~~ 12 sec. w/SIS, ≥ ~~270~~ 6.5 sec. and ≤ 300 sec. wo/SIS	NA
10.b. EDG Start on LOOP	1,2,3,4,(k~~m~~)	NA	≥ ~~4830~~ 4692 V and ≤ ~~4970~~ 5085 V; ≥ 0.~~4~~ 17 sec. and ≤ 7~~0.6~~ sec.	NA
11.a. Chemical and Volume Control System (CVCS) Charging Line Isolation on High-High Pressurizer Level	1,2,3	3 divisions	≤ ~~80~~88% Measuring Range	M
11.b. CVCS Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating)	5[(l~~n~~)],6	3 divisions	~~927 ppm~~(b)	P
11.c. CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions	3,4[(~~o~~m)],5[(m~~o~~)]	3 divisions	(b~~d~~)	P

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

~~(b) If the as-found setpoint is outside its predefined as-found tolerance, then the Trip/Actuation Function shall be evaluated to verify that it is functioning as required before returning the Trip/Actuation Function to service.~~

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(b~~d~~) As specified in the COLR.

(k~~m~~) When associated EDG is required to be OPERABLE by LCO 3.8.2.

(l~~n~~) With two or less RCPs in operation.

(m~~o~~) With three or more RCPs in operation.

Table 3.3.1-2 (page 6 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[a]	SETTING BASIS~~LIMITING TRIP SETPOINT[b][c]~~	CONDITION
12.a. Pressurizer Safety Relief Valve (PSRV) Actuation - First Valve	(n~~p~~)	3 divisions	(o~~q~~)	Q
12.b. PSRV Actuation - Second Valve	(n~~p~~)	3 divisions	(o~~q~~)	Q
13. Control Room Heating, Ventilation, and Air Conditioning Reconfiguration to Recirculation Mode on High Intake Activity	1,2,3,4	3 divisions	≤ 3 x background	N
	5,6,(p~~r~~)	3 divisions	≤ 3 x background	R

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

~~(b) If the as-found setpoint is outside its predefined as-found tolerance, then the Trip/Actuation Function shall be evaluated to verify that it is functioning as required before returning the Trip/Actuation Function to service.~~

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(n~~p~~) When the PSRVs are required to be OPERABLE by LCO 3.4.11.

(o~~q~~) The LTOP arming temperature is specified in the PTLR.

(p~~r~~) During movement of irradiated fuel assemblies.

~~[Reviewers Note: The values specified in brackets in the Limiting Trip Setpoint column are included for reviewer information only. A plant-specific setpoint study will be conducted. The values in Limiting Trip Setpoint column will then be replaced after the completion of this study.]~~

3.3 INSTRUMENTATION

3.3.2 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.2 The PAM instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

--NOTE--

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required division inoperable.	A.1 Restore required division to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.5.	Immediately
C. One or more Functions with two required division inoperable.	C.1 Restore one division to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

---NOTE---
This SR applies to each PAM instrumentation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CALIBRATION	24 months
SR 3.3.2.2	Perform SENSOR OPERATIONAL TEST of the Safety Information and Control System division performing the PAM functions listed in Table 3.3.2-1.	24 months

Table 3.3.2-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED NUMBER OF DIVISIONS
1.	Cold Leg Temperature (Wide Range)	1 per loop
2.	Containment Isolation Valve Position Indication	2(a)(b)
3.	Containment Pressure	2
4.	Emergency Feedwater Storage Pool Level	1 per pool
5.	Emergency Feedwater System Flow	1 per loop
6.	Extra Boration System Flow	2
7.	Hot Leg Injection Flow	1 per loop
8.	Hot Leg Pressure (Wide Range)	1 per loop
9.	Hot Leg Temperature (Wide Range)	1 per loop
10.	In-containment Refueling Water Storage Tank Level	2
11.	Incore Temperature	2 per quadrant
12.	Power Range Monitors	2
13.	Pressurizer Level	2
14.	Radiation Monitor - Containment High Range	2
15.	Radiation Monitor - Main Steam Line Activity	1 per line
16.	Source Range Monitors	2
17.	Steam Generator Level (Wide Range)	2 per SG
18.	Steam Generator Pressure	2 per SG

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication division is required for penetration flow paths with only one installed control room indication division.

3.3 INSTRUMENTATION

3.3.3 Remote Shutdown System (RSS)

LCO 3.3.3 The RSS Functions shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS

---NOTE---

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable.	A.1 Restore required Functions to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Verify each required control circuit and transfer switch is capable of performing the intended function.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.3.2	-------------------------------NOTE------------------------------ Neutron detectors are excluded from the CALIBRATION. --- Perform CALIBRATION for each required instrument division.	24 months
SR 3.3.3.3	Perform SENSOR OPERATIONAL TEST of each required Safety Information and Control System division performing the Remote Shutdown System functions.	24 months

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Two MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSSV inoperable.	A.1 Verify associated Main Steam Relief Train is OPERABLE.	Immediately
	AND	
	A.2 Restore MSSV to OPERABLE status.	30 days
B. Two MSSVs inoperable.	B.1 Restore one MSSV to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition A or B not met. OR Three or more MSSVs inoperable.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify, for each steam generator, one MSSV lift setpoint of ≥ 1416.2 psig and ≤ 1503.8 psig and other MSSV lift setpoint of ≥ 1445.3 psig and ≤ 1534.7 psig in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ± 1%.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Four MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSIVs inoperable due to one associated control line inoperable in MODE 1.	A.1 Restore control line(s) to OPERABLE status.	72 hours
B. One MSIV is inoperable in MODE 1 for reasons other than Condition A.	B.1 Restore MSIV to OPERABLE status.	8 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 2.	6 hours
D. ------------NOTE------------ Separate Condition entry is allowed for each MSIV. --------------------------------- One or more MSIVs inoperable in MODE 2 or 3.	D.1 Close MSIV. <u>AND</u> D.2 Verify MSIV is closed.	8 hours Once per 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Cycle each MSIV pilot valve.	31 days
SR 3.7.2.2	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. -- Perform a partial closure test of each MSIV.	92 days
SR 3.7.2.3	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. -- Verify the isolation time of each MSIV is within limits.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.2.4	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	24 months on a STAGGERED TEST BASIS for each MSIV pilot valve

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater (MFW) Valves

LCO 3.7.3 Four MFW Full Load Isolation Valves (MFWFLIVs), MFW Full Load Control Valves (MFWFLCVs), MFW Low Load Isolation Valves (MFWLLIVs), MFW Low Load Control Valves (MFWLLCVs), MFW Very Low Load Control Valves (MFWVLLCVs), and MFW Main Isolation Valves (MFWMIVs) shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MFWs are closed.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each MFW flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more full load flow paths with one MFW valve inoperable.	A.1 Restore MFW valve to OPERABLE status.	7 days
B. One or more full load flow paths with two MFW valves inoperable.	B.1 Restore one MFW valve to OPERABLE status.	72 hours
C. One or more full load flow paths with three MFW valves inoperable.	C.1 Restore one MFW valve to OPERABLE status.	8 hours
D. One or more low load or very low load flow paths with one or more MFWLLIV, MFWLLCV, or MFWVLLCV valves inoperable.	D.1 Isolate associated flow path. AND D.2 Verify the flow path is isolated.	8 hours 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the isolation time of each MFWFLIV, MFWFLCV, MFWLLIV, MFWLLCV, MFWVLLCV, and MFWMIV is within limits.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify each MFWFLIV, MFWFLCV, MFWLLIV, MFWLLCV, MFWVLLCV, and MFWMIV actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.4 Main Steam Relief Trains (MSRTs)

LCO 3.7.4 Four MSRTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSRIVs inoperable due to one or both pilot valves in control line inoperable for opening. OR One or more required MSRIVs inoperable due to one pilot valve open in one or both control lines.	A.1 Restore pilot valve(s) to OPERABLE status.	30 days
B. One or two MSRIVs inoperable for opening. OR One or two MSRIVs inoperable for closing. OR One or two MSRCVs inoperable.	B.1 Verify associated main steam safety valve(s) are OPERABLE. AND B.2 Restore valve(s) to OPERABLE status.	Immediately 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Times of Condition A or B not met. <u>OR</u> Three or more required MSRIVs inoperable for opening. <u>OR</u> Three or more required MSRVs inoperable.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each MSRIV.	24 months on STAGGERED TEST BASIS for each control line
SR 3.7.4.2	Verify one complete cycle of each MSRCV.	24 months
SR 3.7.4.3	Verify each MSRIV automatically actuates on an actual or simulated steam pressure setpoints.	24 months
SR 3.7.4.4	Verify each MSRCV is automatically positioned on an actual or simulated actuation signal.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.4.5	Verify each MSRCV is automatically switched into steam generator pressure control mode on an actual or simulated MSRIV opening signal.	24 months

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Four EFW trains shall be OPERABLE.

---NOTE---
Only one EFW train is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

--NOTE--
LCO 3.0.4.b is not applicable for two or more EFW trains inoperable when entering MODE 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One EFW train inoperable in MODE 1, 2, or 3.	A.1 Restore EFW train to OPERABLE status.	120 days
B. Two EFW trains inoperable in MODE 1, 2, or 3.	B.1 Restore one EFW train to OPERABLE status.	72 hours
C. Required Action and associated Completion Time for Condition A or B not met. <u>OR</u> Three EFW trains inoperable in MODE 1, 2, or 3.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	6 hours 12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Four of the EFW trains inoperable in MODE 1, 2, or 3.	D.1	------------NOTE---------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. ------------------------------------	
		Initiate action to restore one EFW train to OPERABLE status.	Immediately
E. One required EFW train inoperable in MODE 4.	E.1	Initiate action to restore required EFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	Verify EFW pump suction and supply header isolation valves are locked open.	31 days
SR 3.7.5.3	Cycle each EFW discharge header cross-connect valve.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.4	Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.5	Verify, on an actual or simulated actuation signal, each EFW automatic valve that is not locked, sealed, or otherwise secured in position actuates to the correct position, each EFW pump starts automatically, and flow rate is controlled within required limits and steam generator level is controlled within limits.	24 months
SR 3.7.5.6	Verify proper alignment of the required EFW flow paths by verifying flow from the EFW storage pool to its respective steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days

3.7 PLANT SYSTEMS

3.7.6 Emergency Feedwater (EFW) Storage Pools

LCO 3.7.6 Four EFW Storage Pools shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One EFW Storage Pool inoperable.	A.1 Verify the usable volume in the three remaining EFW Storage Pools is ≥ 300,000 gal. <u>AND</u> A.2 Declare associated EFW train inoperable.	Immediately Immediately
B. Two or more EFW Storage Pools inoperable. <u>OR</u> Usable volume in EFW Storage Pools < 300,000 gal.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4, without reliance on steam generator for heat removal.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the EFW Storage Pools contain a usable volume ≥ 300,000 gal.	24 hours
SR 3.7.6.2	Verify each EFW Storage Pool supply cross connect valve is locked open.	31 days

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Four CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTE--
Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW System.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCW train inoperable.	A.1 ---------------NOTE------------- Required Action A.1 is not applicable if CCW trains are inoperable in both CCW headers supplying the reactor coolant pump (RCP) thermal barrier cooling common loop and Condition B is entered. -------------------------------------	
	Align RCP thermal barrier cooling common loop to the CCW header with two OPERABLE CCW trains.	72 hours
	<u>AND</u>	
	A.2 Restore CCW train to OPERABLE status.	120 days
B. Two CCW trains inoperable.	B.1 Restore one CCW train to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	------------------------------NOTE----------------------------- Isolation of CCW flow to individual components, other than the RCP thermal barrier cooling common loop, does not render the CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.8 Essential Service Water (ESW) System

LCO 3.7.8 Four ESW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTES---

1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generators made inoperable by ESW System.

2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loop - MODE 4," for residual heat removal loops made inoperable by ESW System.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ESW train inoperable.	A.1 Restore ESW train to OPERABLE status.	120 days
B. Two ESW trains inoperable.	B.1 Restore one ESW train to OPERABLE status.	72 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify water level of each ESW cooling tower basin is ≥ 27.2 feet.	24 hours
SR 3.7.8.2	Verify water temperature of each ESW cooling tower basin is ≤ 90°F.	24 hours
SR 3.7.8.3	-------------------------------NOTE------------------------------ Isolation of ESW flow to individual components does not render the ESW System inoperable. --- Verify each ESW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.8.4	Operate each ESW cooling tower fan for ≥ 15 minutes in all speed settings.	31 days
SR 3.7.8.5	Verify each ESW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.8.6	Verify each ESW pump and cooling tower fan starts automatically on an actual or simulated actuation signal.	24 months
SR 3.7.8.7	Verify the ability to supply makeup water to each ESW basin at ≥ 300 gpm.	24 months

3.7 PLANT SYSTEMS

3.7.9 Safety Chilled Water (SCW) System

LCO 3.7.9 Four SCW trains shall be OPERABLE and in operation.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---

Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by SCW System.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SCW train inoperable or not in operation.	A.1 Restore SCW train to OPERABLE status and in operation.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify each SCW train is in operation.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.9.2	-------------------------------NOTE------------------------------ Isolation of SCW flow to individual components does not render the SCW System inoperable. --- Verify each SCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.9.3	Verify each SCW train has the capability to remove the design heat load.	24 months
SR 3.7.9.4	Verify, on an actual or simulated loss of offsite power signal, each SCW train restarts following re-energization of the associated AC electrical power division.	24 months

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Filtration (CREF)

LCO 3.7.10 Two CREF trains shall be OPERABLE.

--NOTE--
The control room envelope (CRE) may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREF train inoperable.	A.1 Restore CREF train to OPERABLE status.	7 days
B. Two CREF trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4	B.1 Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>	
	B.2 Verify mitigating actions ensure CRE occupant exposures to radiological~~,~~ [[~~chemical~~]], and smoke hazards will not exceed limits.	24 hours
	<u>AND</u>	
	B.3 Restore CRE boundary to OPERABLE status.	90 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	~~--------REVIEWER'S NOTE---------~~ ~~The need for the toxic gas isolation state will be determined by the COL Applicant.~~ ~~---~~ D.1 ~~-------------NOTE-------------~~ ~~Place CREF train in toxic gas isolation state if automatic transfer to toxic gas isolation state is inoperable.~~ ~~---------------------------------~~ Place OPERABLE CREF train in emergency mode. <u>OR</u> D.2 Suspend movement of irradiated fuel assemblies.	 Immediately Immediately
E. Two CREF trains inoperablein MODE 5 or 6, or during movement of irradiated fuel assemblies.	E.1 Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CREF trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREF train for ≥ 15 minutes with the heaters operating.	31 days
SR 3.7.10.2	Perform required CREF train testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.10.3	Verify each CREF train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

3.7 PLANT SYSTEMS

3.7.11 Control Room Air Conditioning System (CRACS)

LCO 3.7.11 Four CRACS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two CRACS trains inoperable.	A.1 Restore the inoperable CRACS train(s) to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1 Place OPERABLE CRACS train in operation. OR C.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
D. Three or more CRACS trains inoperable in MODE 5 or 6 or during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Three or more CRACS trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify each CRACS train has the capability to remove the design heat load.	24 months

3.7 PLANT SYSTEMS

3.7.12 Safeguard Building Controlled Area Ventilation System (SBVS)

LCO 3.7.12 Two SBVS Accident Exhaust Filtration trains shall be OPERABLE.

---NOTE---
The safeguards building and fuel building boundaries may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SBVS Accident Exhaust Filtration train inoperable.	A.1 Restore SBVS Accident Exhaust Filtration train to OPERABLE status.	7 days
B. Two SBVS Accident Exhaust trains inoperable due to inoperable safeguards building or fuel building boundary.	B.1 Restore safeguards building and fuel building boundaries to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Two SBVS Accident Exhaust Filtration train inoperable for reasons other than Condition B.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify safeguards building and fuel building negative pressure is ≥ 0.25 inches water gauge.	12 hours
SR 3.7.12.2	Verify each safeguards building and fuel building access door is closed, except when the access opening is being used for entry and exit.	31 days
SR 3.7.12.3	Operate each SBVS Accident Exhaust Filtration train for ≥ 15 minutes with the heaters operating.	31 days
SR 3.7.12.4	Perform required SBVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.5	Verify each SBVS Accident Exhaust Filtration train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.12.6	Verify the safeguards building and fuel building can be drawn down to a negative pressure ≥ 0.25 inches water gauge in ≤ 305 seconds after a start signal using one SBVS Accident Exhaust Filtration train.	24 months on a STAGGERED TEST BASIS for each SBVS Accident Exhaust Filtration train
SR 3.7.12.7	Verify the safeguards building and fuel building can be maintained at a negative pressure ≥ 0.25 inches water gauge using one SBVS Accident Exhaust Filtration train at a flow rate of ≤ 2640 cfm.	24 months on a STAGGERED TEST BASIS for each SBVS Accident Exhaust Filtration train

3.7 PLANT SYSTEMS

3.7.13 Safeguards Building Ventilation System Electrical Division (SBVSED)

LCO 3.7.13 Four SBVSED trains shall be OPERABLE and in operation.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SBVSED train inoperable or not in operation.	A.1 Restore SBVSED train to OPERABLE status and in operation.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify each SBVSED train is in operation.	24 hours
SR 3.7.13.2	Verify each SBVSED train has the capability to remove the design heat load.	24 months
SR 3.7.13.3	Verify, on an actual or simulated loss of offsite power signal, each SBVSED train restarts following re-energization of the associated AC electrical power division.	24 months

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Storage Pool Water Level

LCO 3.7.14 The spent fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1	--------------NOTE-------------- LCO 3.0.3 is not applicable. ------------------------------------- Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the spent fuel storage pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Storage Pool Boron Concentration

~~---Reviewer's Note---~~
~~The design of the spent fuel storage racks is to be provided by the COLA applicant. The required boron concentration will be provided as a part of the spent fuel rack design.~~
~~---~~

LCO 3.7.15 The spent fuel storage pool boron concentration shall be ≥ 500 ~~[1291]~~ ppm and boron enrichment shall be ≥ 37%.

APPLICABILITY: When fuel assemblies are stored in the spent fuel storage pool and a spent fuel storage pool verification has not been performed since the last movement of fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool boron concentration or enrichment not within limit.	------------------NOTE------------------ LCO 3.0.3 is not applicable. --	
	A.1 Suspend movement of fuel assemblies in the spent fuel storage pool.	Immediately
	AND	
	A.2.1 Initiate action to restore fuel storage pool boron concentration and enrichment to within limits.	Immediately
	OR	
	A.2.2 Initiate action to perform a spent fuel storage pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel storage pool boron concentration is within limit.	7 days
SR 3.7.15.2	Verify the isotopic concentration of B^{10} in the spent fuel storage pool is within limit.	24 months

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Storage

LCO 3.7.16 ~~Fuel shall be stored in approved locations in the spent fuel storage pool~~The combination of initial enrichment and burnup of each fuel assembly stored in Region 2 of the spent fuel storage pool shall be within the limits specified in Figure 3.7.16-1.

~~--Reviewer's Note--~~
~~The design of the spent fuel storage racks is to be provided by the COLA applicant. The required spent fuel storage configuration will be provided as a part of the spent fuel rack design.~~
~~--~~

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 of the spent fuel storage ~~racks~~pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 --------------NOTE-------------- LCO 3.0.3 is not applicable. -------------------------------------	
	Initiate action to ~~restore fuel storage to within requirements~~move the non-complying fuel assembly to an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means ~~that fuel assemblies are stored in approved locations~~the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1.	Prior to storing the fuel assembly in Region 2 of the spent fuel storage

SURVEILLANCE	FREQUENCY
	pool

[

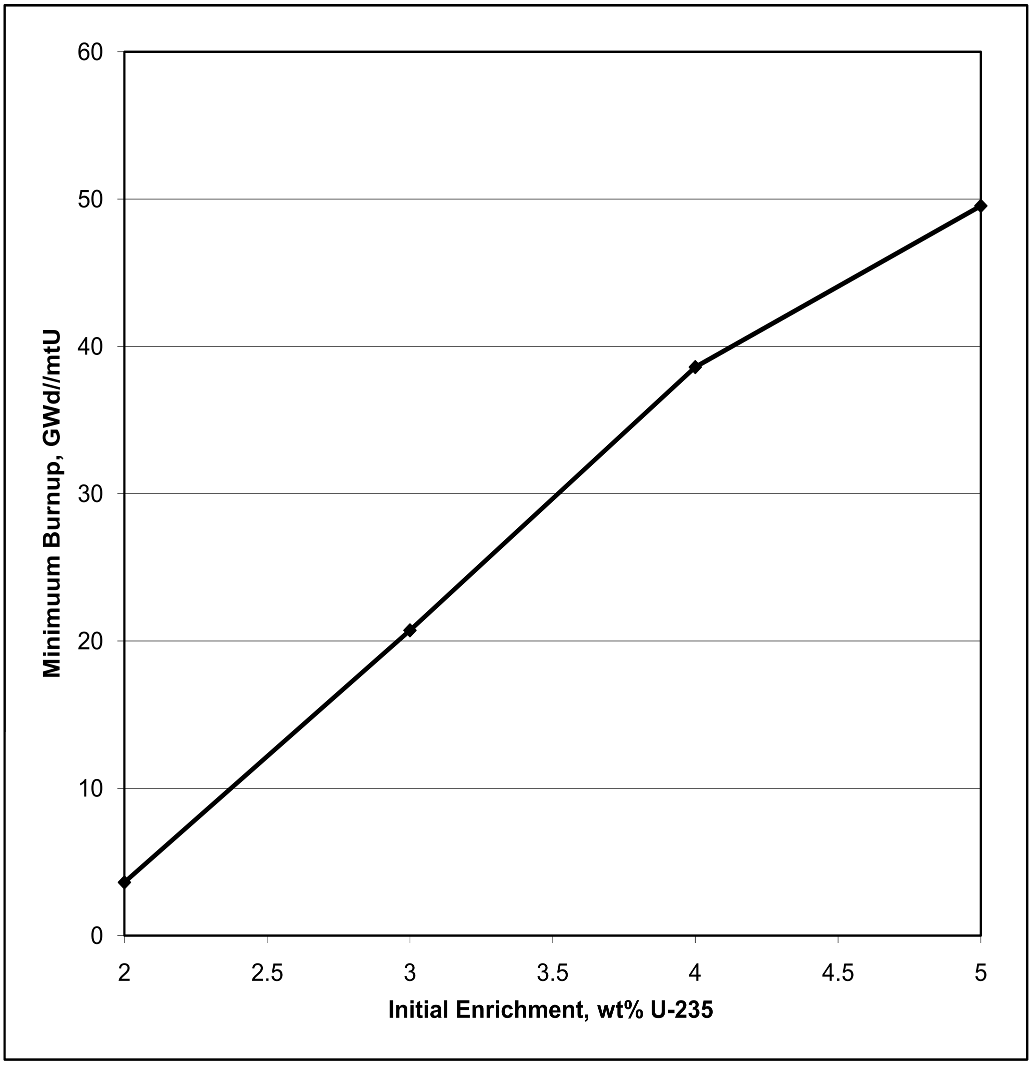

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Figure 3.7.16-1 (page 1 of 1)

Fuel Assembly Burnup Requirements for Region 2

3.7 PLANT SYSTEMS

3.7.17 Secondary Specific Activity

LCO 3.7.17 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.	31 days

4.1 Site Location

~~[A COL Applicant that references the U.S. EPR design certification will provide site-specific information for Section 4.1, Site Location.]~~ The site for the Nine Mile Point 3 Nuclear Power Plant (NMP3NPP) is located in the north sector of Oswego County and is along the south shore of Lake Ontario, about 6 miles east of Oswego, NY. The metropolitan centers closest to the NMP3NPP site are Syracuse, NY, approximately 35 miles (56 km) to the south; Rochester, NY, approximately 70 miles (113 km) to the southwest; Buffalo, NY, approximately 140 miles (225 km) to the southwest; Scranton, PA, approximately 170 miles (274 km) to the south; Albany, NY, approximately 170 miles (274 km) to the southeast; and Toronto, Ontario, Canada, approximately 235 miles (378 km) to the west. The exclusion area boundary for NMP3NPP is a circle with a radius of 2,220 feet, or approximately 0.42 mi except for the Ontario Bible Camp property that is excluded from the EAB.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies. Each assembly shall consist of a matrix of fuel rods clad with a zirconium based alloy with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 89 control rod assemblies. The control material shall be silver indium cadmium as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[~~FSAR~~]~~ Section 9.1; ~~and~~

c. A nominal ~~[~~10~~1~~.9~~28]~~ inch center to center distance between fuel assemblies placed in ~~the spent fuel storage racks~~Region 1 and a nominal 9.028 inch center to center distance between fuel assemblies placed in Region 2 of the spent fuel storage racks;~~.~~

4.3 Fuel Storage (continued)

d. New or partially spent fuel assemblies with any discharge burnup may be allowed unrestricted storage in Region 1 of Figure 4.3-1;

e. Partially spent fuel assemblies meeting the initial enrichment and burnup requirements of LCO 3.7.16, “Spent Fuel Storage,” may be stored in Region 2 of Figure 4.3-1.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[~~FSAR~~]~~ Section 9.1;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in ~~[~~FSAR~~]~~ Section 9.1; and

d. A nominal ~~[~~[10~~1~~.9]~~28]~~ inch center to center distance between fuel assemblies placed in the new fuel storage racks.

~~------------------------------------REVIEWER’S NOTE------------------------------------~~
~~Storage rack uncertainties are discussed in the FSAR or COLA Section 9.1~~
~~--~~

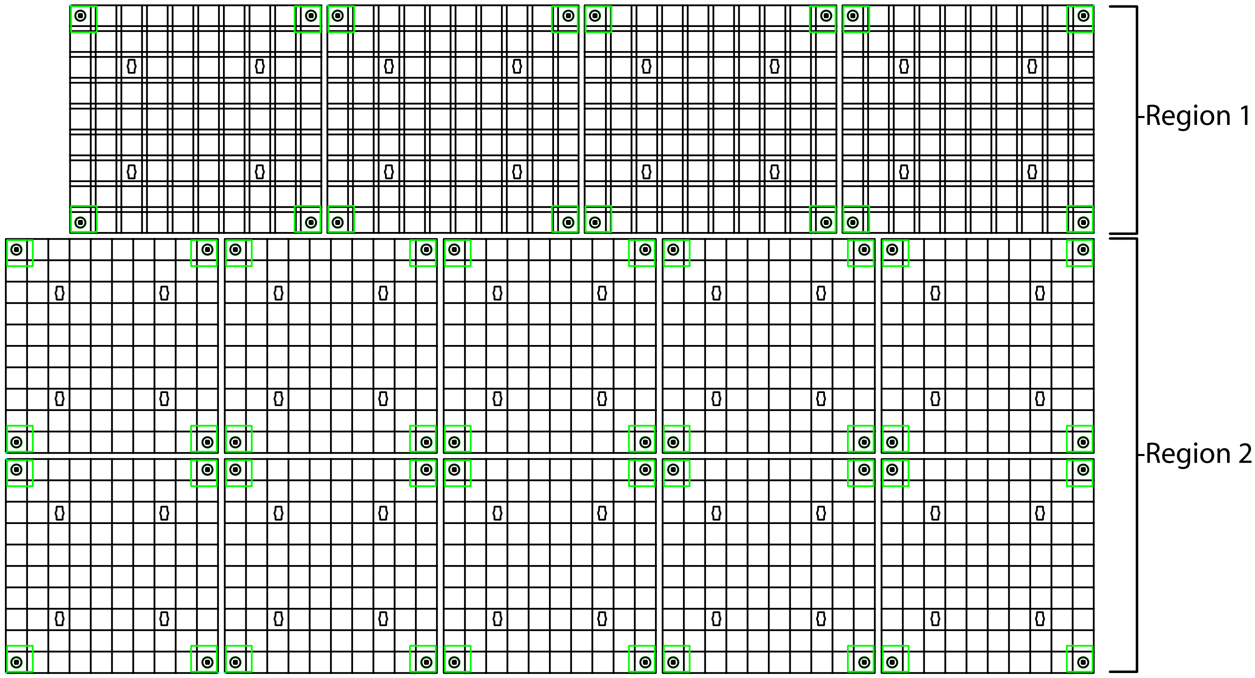
4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 23 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than [~~1121~~1360~~]~~ fuel assemblies.

[



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~~[~~Region 1 (Racks 1~~1~~ through ~~1~~4) – 360 locations

Region 2 (Racks 5~~1~~ through 15~~0~~) – 1000 locations

Total Storage Locations – 1360~~]~~

Figure 4.3-1 (page 1 of 1)

Discrete Two Region Spent Fuel Pool Rack Layout

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

--REVIEWER'S NOTE--------------------------------------

~~1.Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adoptable to those unit staffs requiring special titles because of unique organizational structures~~Organizational positions listed or described in the Administrative Controls section shall have corresponding plant-specific titles specified in the Final Safety Analysis Report.

~~1.The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title applied with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.~~

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 The shift supervisor shall be responsible for the control room command function. During any absence of the shift supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Operator license shall be designated to assume the control room command function. During any absence of the shift supervisor from the control room while the unit is in MODE 5 or 6, an individual with an active Senior Operator license or Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the FSAR/QA Plan.

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

~~------------------------------------REVIEWER'S NOTE------------------------------------~~
~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~
~~--~~

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODE 1, 2, 3, or 4;

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements;

5.2 Organization

5.2.2 Unit Staff (continued)

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position;

d. Administrative controls shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

Any deviation from the above guidelines shall be authorized in advance by the plant manager or designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned;

e. The operations manager or assistant operations manager shall hold a Senior Operator license; and

f. When the reactor is operating in MODE 1, 2, 3, or 4, an individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

~~---REVIEWER'S NOTE---~~
~~Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~
~~--~~

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 3, 2000, with the following exception:

a. During cold license operator training prior to Commercial Operation, the following Regulatory Position C.1.b of Regulatory Guide 1.8, Revision 2, 1987, applies:

Cold license operator candidates meet the training elements defined in ANS/ANSI 3.1-1993 but are exempt from the experience requirements defined in ANS/ANSI 3.1-1993.

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed Operator are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program, and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.1 and Specification 5.6.2.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s); and

   b) A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after the approval of the plant manager; and

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Low Head Safety Injection, Medium Head Safety Injection, and Nuclear Sampling. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system once per 24 months.

The provisions of SR 3.0.2 are applicable.

5.5.3 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

5.5 Programs and Manuals

5.5.3 Radioactive Effluent Controls Program (continued)

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin; and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance Frequencies.

5.5.4 Component Cyclic or Transient Limit

This program provides controls to track the FSAR Section 3.9.1.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5 Programs and Manuals

5.5.5 Containment Post Tensioning Surveillance Program

This program provides for the monitoring of the containment post tensioning force over time. Tendons used in the containment structure are fully grouted and the structure itself is not exposed to the environment during its operational life. The program shall include initial base line measurements prior to initial operation. The Containment Post Tensioning Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section III Division 2 of the ASME Boiler and Pressure Vessel Code, 2004, and Regulatory Guide 1.90, Rev. 1. Since the U.S. EPR has no ungrouted tendons, force monitoring of ungrouted tendons is not required.

The provisions of SR 3.0.3 are applicable to the Containment Post Tensioning Surveillance Program inspection frequencies.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

5.5 Programs and Manuals

5.5.7 Inservice Testing Program (continued)

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

    1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.12, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain indications with a depth equal to or exceeding 40% of the nominal tube wall thickness per eddy current results shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of Specifications 5.5.8.d.1, 5.5.8.d.2, and 5.5.8.d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems in accordance with Regulatory Guide 1.52, Revision 3, ASME N510-1989, and ASME AG-1-2003. The frequencies of 5.5.10a, 5.5.10b and 5.5.10c are in accordance with Regulatory Guide 1.52, Revision 3. The frequency for 5.5.10d and 5.5.10e is 24 months.

a. Demonstrate for each of the ESF systems that an in place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
Annulus Ventilation System (AVS)	≥ 1060 and ≤ 1295
Safeguards Building Controlled Area Ventilation System (SBCAVS)	≥ 2160 and ≤ 2640
Control Room Emergency Filtration (CREF)	≥ 3600 and ≤ 4400
Containment Low Flow Purge Subsystem (CLFPS)	≥ 2700 and ≤ 3300

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
AVS	≥ 1060 and ≤ 1295
SBCAVS	≥ 2160 and ≤ 2640
CREF	≥ 3600 and ≤ 4400
CLFPS	≥ 2700 and ≤ 3300

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 3, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below.

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Penetration	RH	Face Velocity (fpm)
AVS	0.5%	≤ 70%	300
SBCAVS	0.5%	≤ 70%	375
CREF	0.5%	≤ 70%	250
CLFPS	0.5%	≤ 70%	375

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at the system flowrate specified below..

ESF Ventilation System	Delta P (in wg)	Flowrate (cfm)
AVS	7.5	≥ 1060 and ≤ 1295
SBCAVS	7.5	≥ 2160 and ≤ 2640
CREF	7.5	≥ 3600 and ≤ 4400
CLFPS	7.5	≥ 2700 and ≤ 3300

e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF Ventilation System	Wattage (kw)
AVS	≥ 4 and ≤ 8
SBCAVS	≥ 9 and ≤ 13
CREF	
Outside Air	≥ 30 and ≤ 38
Emergency Filter Bank	≥ 13 and ≤ 17
CLFPS	≥ 12 and ≤ 16

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP Test Frequencies.

5.5 Programs and Manuals

5.5.11 Gaseous Waste Processing System Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste Processing System and the quantity of radioactivity contained in gas delay beds. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Waste Processing System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in the gas delay beds is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the beds' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

~~---REVIEWER'S NOTE---~~
~~The U.S. EPR does not have outdoor liquid radwaste tanks. If a COL Applicant adds outdoor liquid radwaste tanks, this program will be modified accordingly.~~
~~---~~

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity or an absolute specific gravity within limits;

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil; and

5.5 Programs and Manuals

5.5.12 Diesel Fuel Oil Testing Program (continued)

3. A clear and bright appearance with proper color, or a water and sediment content within limits.

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in Specification 5.5.12.a, above, are within limits for ASTM 2D fuel oil, and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program Surveillance Frequencies.

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior regulatory approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.13.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

   1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

   2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

   3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

   4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

   1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

   2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

   3. A required system redundant to the support system(s) for the supported systems described in Specifications 5.5.14.b.1 and 5.5.14.b.2 above is also inoperable.

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the approved exceptions.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident is 52.0 psig. P_a is conservatively assumed to be 55 psig. The containment design pressure is 62 psig.

c. The maximum allowable containment leakage rate, La, at Pa, shall be 0.25% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

b) For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 10 psig.

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

~~-----REVIEWER'S NOTE-----~~
~~As discussed in FSAR Section 6.2.6, the U.S. EPR has no penetrations that are classified as bypass leakage paths.~~
~~----------~~

5.5.16 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer including the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plate.

5.5.17 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filtration System (CREFS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event~~, [[hazardous chemical release,]]~~ or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary;

b. Requirements for maintaining CRE boundary in its design condition including configuration control and preventive maintenance;

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0;

5.5 Programs and Manuals

5.5.17 Control Room Envelope Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization MODE of operation by one train of the CREFS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary;

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in Specification 5.5.17.c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences~~. Unfiltered air inleakage limits for [[hazardous chemicals]] must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis~~; and

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by Specifications 5.5.17.c and 5.5.17.d, respectively.

5.5.18 Setpoint Control Program (SCP)

a. The Setpoint Control Program shall document the Limiting Trip Setpoints (LTSPs), Nominal Trip Setpoints (NTSPs) (where desired), Allowable Values (AVs), and As-Found and As-Left Tolerance Bands for each of the required Technical Specification Instrumentation Functions in Specification 3.3.1, "Protection System (PS)."

b. The analytical methods used to determine the LTSPs, NTSPs (if applicable), AVs and As-Found Tolerance and As-Left Tolerance Bands shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. ANP-10275P-A, "U.S. EPR Instrument Setpoint Methodology Topical Report," ~~March 2007~~February 2008; and

2. ANP-10287, "Incore Trip Setpoint and Transient Methodology for U.S. EPR," November 2007.

5.5 Programs and Manuals

5.5.18 Setpoint Control Program (SCP) (continued)

c. The Setpoint Control Program shall also contain the following:

1. Provisions for evaluation of an instrumentation division to verify it is functioning as required, before return to service, if the as-found division setpoint is found to be conservative with respect to its AV, but outside its predefined As-Found Tolerance Band; and

2. Provisions for resetting an instrumentation division setpoint to a value that is within the As-Left Tolerance Band of the associated LTSP, or within the As-Left Tolerance Band of the associated NTSP (if applicable), or otherwise declaring the instrument division inoperable.

d. The Setpoint Control Program, including any revisions or supplements, shall be provided to the NRC upon issuance:

1. Prior to initial fuel load; and
2. On a frequency consistent with 10 CFR 50.71(e).

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Annual Radiological Environmental Operating Report

~~-------------------------------------REVIEWER'S NOTE-------------------------------------~~
~~[A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.]~~
~~---~~

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.2 Radioactive Effluent Release Report

~~-------------------------------------REVIEWER'S NOTE-------------------------------------~~
~~[A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.]~~
~~---~~

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each cycle, or prior to any remaining portion of a cycle, and shall be documented in the COLR for the following:

1. LCO 2.1.1, "Reactor Core SLs";
2. LCO 3.1.1, "Shutdown Margin (SDM)";
3. LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
4. LCO 3.1.5, "Shutdown Bank Insertion Limits";
5. LCO 3.1.6, "Control Bank Insertion Limits";
6. LCO 3.1.7, "Rod Control Cluster Assembly (RCCA) Position Indication";
7. LCO 3.1.9, "Physics Test Exceptions – MODE 2";
8. LCO 3.2.1, "Linear Power Density";
9. LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
10. LCO 3.2.3, "Departure From Nucleate Boiling Ratio (DNBR)";
11. LCO 3.2.4, "Axial Offset";
12. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Ratio (DNB) Limits"; and
13. LCO 3.9.1, "Boron Concentration During Refueling Operations."

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. ANP-10263P, "Codes and Methods Applicability Report for the U.S. EPR";

2. DOE/ET/34212-41 BAW-1810, "Urania Gadolinia: Nuclear Model Development and Critical Experiment Benchmark," April 1984;

3. EMF-96-029(P)(A) Volume 1 and 2, "Reactor Analysis Systems for PWRs," January 1997;

4. BAW-10221P-A, "NEMO-K, A Kinetics Solution in NEMO", September 1998;

5. ANSI/ANS 19.6.1-2005, "Reload Startup Physics Tests for Pressurized Water Reactors," American Nuclear Society, 2005;

6. BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code," June 2002; and

7. BAW-10163P-A, "Core Operating Limit Methods for Westinghouse Designed PWRs," June 1989.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined assuming operation at RATED THERMAL POWER such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, low temperature overpressure protection settings, and pressurizer safety relief valve lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits;"

2. LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. ANP-10283, Rev. 0, "U.S. EPR Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown."

c. The PTLR shall be provided to the applicable regulatory body upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.2, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6 Reporting Requirements

5.6.6 Containment Post Tensioning Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Containment Post Tensioning Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, "Steam Generator (SG) Program". The report shall include:

a. The scope of inspections performed on each SG;

b. Active degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;

e. Number of tubes plugged during the inspection outage for each active degradation mechanism;

f. Total number and percentage of tubes plugged to date;

g. The results of condition monitoring, including the results of tube pulls and insitu testing; and

h. The plugging percentage for all plugging in each SG.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes Specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess one of the following:

    1. A radiation monitoring device that continuously displays radiation dose rates in the area;

    2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint;

    3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

    4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source of from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door, gate, or other barrier that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designees; and

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes Specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

5.7 High Radiation Area

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source of from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation</u> (continued)

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess one of the following:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint;

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where Specifications 5.7.2.d.2 and 5.7.2.d.3 above are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displaces radiation dose rates in the area.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source of from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

B 3.3 INSTRUMENTATION

B 3.3.1 Protection System (PS)

BASES

BACKGROUND

The PS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary during anticipated operational occurrences (AOOs). The PS also initiates the Engineered Safety Features (ESF) actuations that are used to mitigating accidents. The ESF actuates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits, maintain the Reactor Coolant System (RCS) pressure boundary, and mitigate the consequences of accidents that could result in potential exposures comparable to the guidelines set forth in 10 CFR 100 during AOOs and ensures acceptable consequences during accidents.

The PS initiates and the Safety Automation System (SAS) controls the necessary safety systems to protect against violating core design limits, maintain the RCS pressure boundary, and mitigate the consequences of accidents that could result in potential exposures comparable to the guidelines set forth in 10 CFR 100 during anticipated operational occurrences and ensures acceptable consequences during postulated accidents.

The four redundant divisions of the PS are physically separated in their respective safeguard buildings. The four divisionally separated rooms containing the PS equipment are in different fire zones. Therefore, in general, the consequences of internal hazards (e.g., fire), would impact only one PS division.

The PS architecture is four-fold redundant for both reactor trip and ESF functions. A single failure during corrective or periodic maintenance, or a single failure and the effects of an internal hazard does not prevent performance of the safety functions. For the reactor trip functions, each PS division actuates one division of the reactor trip devices based on redundant processing performed in four divisions. For ESF functions, the redundancy of the safety function as a whole is defined by the redundancy of the ESF system mechanical trains. In general, this results in one PS division actuating one mechanical train of an ESF system based on redundant processing performed in four divisions. The PS not only supports the redundancy of the mechanical trains, but also enhances this redundancy through techniques such as redundant actuation voting.

BASES

BACKGROUND (continued)

Three of the four divisions are necessary to meet the redundancy and testability of GDC 21 in 10 CFR 50, Appendix A (Ref. 3). The fourth division provides additional flexibility by allowing one division to be removed from service for maintenance or testing while still maintaining a minimum two-out-of-three logic. Thus, even with a division inoperable, no single additional failure in the PS can either cause an inadvertent trip/ESF or prevent a required trip/ESF from occurring.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the PS, as well as LCOs on other reactor system parameters and equipment performance. ~~The subset of LSSS that directly protect against violating the reactor core and RCS pressure boundary safety limits during AOOs are referred to as Safety Limit LSSS (SL-LSSS).~~

Technical Specifications are required by 10 CFR 50.36 to contain LSSS defined by the regulation as "...settings for automatic protective devices...so chosen that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protective devices must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur. When LSSS is specified for a variable having a significant safety function, but which does not protect SLs, the setting must be chosen such that automatic protective actions will initiate consistent with the design basis. The Design Limit is the limit of the process variable at which the safety action is initiated to ensure that these automatic protective devices will perform their specified safety function. These limits (i.e., the Analytical Limits and Design Limits) constitute the Setting Basis specified in Table 3.3.1-2.

~~------------------------------- REVIEWER'S NOTE -----------------------------------~~
~~The term "Limiting Trip Setpoint (LTSP)" is generic terminology for the setpoint value calculated by means of the plant-specific setpoint methodology documented in a document controlled under 10 CFR 50.59. The term LTSP indicates that no additional margin has been added between the Analytical Limit and the calculated trip setting. Where margin is added between the Analytical Limit and trip setpoint, the term~~

~~Nominal Trip Setpoint is preferred. The trip setpoint (field setting) may be more conservative than the Limiting or Nominal Trip Setpoint.~~

~~Where the LTSP is not included in Table 3.3.1-2 for the purpose of compliance with 10 CFR 50.36, the plant-specific term for the Limiting or Nominal Trip Setpoint must be cited in Note b of Table 3.3.1-2. The brackets indicate plant-specific terms may apply, as reviewed and approved by the NRC. The as-found and as-left tolerances will apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance.~~

BASES

BACKGROUND (continued)

~~Licensees are to insert the name of the document(s) controlled under 10 CFR 50.59 that contain the LTSP and the methodology for calculating the as-left and as-found tolerances, for the phrase "a document controlled under 10 CFR 50.59" in the specifications.~~

~~---~~

The LTSP is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable reaching the Analytical/Design Limit and thus ensuring that the SL would not be exceeded (i.e., for Analytical Limits) or that automatic protective actions occur consistent with the design basis (i.e., for Design Limits). As such, the LTSP accounts for uncertainties in setting the device (e.g., CALIBRATION), uncertainties in how the device might actually perform (e.g., repeatability), changes in the point of action of the device over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the LTSP ensures that SLs are not exceeded and that automatic protective devices will perform their specified safety function. ~~As such, the LTSP meets the definition of a SL-LSSS (Ref. 1).~~

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety function(s)." For automatic protective devices, the required safety function is to ensure that the SL is not exceeded and that automatic protective actions will initiate consistent with design basis. However, use of the LTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protective device setting during a Surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protective device with a setting that has been found to be different from the LTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the LTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded or that automatic protective actions would initiate consistent with the design basis with the "as-found" setting of the protective device. Therefore, the device would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the device to the ~~trip setpoint~~LTSP to account for further drift during the next surveillance interval.

~~However, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. This LTSP specified in Table 3.3.1-2 is the least conservative value of the as-found setpoint that a channel can have during testing such that a channel is OPERABLE if the trip setpoint is found conservative with respect to the Allowable Value during a SENSOR~~

BASES

BACKGROUND (continued)

~~OPERATIONAL TEST (SOT). As such, the Allowable Value differs from the LTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will ensure that an SL is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the setpoint must be left adjusted to a value within the as-left tolerance, and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found). If the actual setting of the device is found to be non-conservative with respect to the Allowable Value, the device would be considered inoperable from a Technical Specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required.~~
However, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. This value is specified in the SCP, as required by Specification 5.5.18, in order to define OPERABILITY of the devices and is designated as the Allowable Value, which is the least conservative value of the as-found setpoint that a division can have during a periodic CALIBRATION or SENSOR OPERATIONAL TEST.

The actual LTSP and Allowable Values (derived for the Setting Basis values specified in Table 3.3.1-2) and the methodology for calculating the as-found and as-left tolerances are maintained in SCP, as required by Specification 5.5.18.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling (DNB),
- Fuel centerline melting shall not occur; and
- The RCS pressure SL of 2803 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within 10 CFR 100 limits. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The PS is segmented into four interconnected modules and associated LCOs for the reactor trips and ESF functions. These modules are:

- Sensors, which include the associated instrumentation;
- Manual actuation switches;

BASES

BACKGROUND (continued)

- Signal Processors, which include:
  - Remote Acquisition Units (RAUs), which acquire the signals from the Self-Powered Neutron Detectors (SPND) and distribute these signals;
  - Acquisition and Processing Units (APUs), which perform calculations and make setpoint comparisons; and
  - Actuation Logic Units (ALUs), which perform voting of the processing results from the redundant APUs in the different divisions and to issue actuation orders based on the voting results; and
- Actuation Devices, which includes the reactor trip breakers and contactors and the Priority Actuation and Control Systems (PACS) control modules for the Reactor Coolant Pump (RCP) bus and trip breakers..

The PS is a digital, integrated reactor protection system and engineered safety features actuation system. Individual sensors, signal processors, or the ALUs that provide the actuation signal voting function, can be associated with multiple reactor trip, ESF functions, and Permissives.

Sensors

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

The Power Density Detector System, which uses SPND and RAUs, provides the in-core monitoring function. The Power Range, Intermediate Range, and Source Range monitors provide the ex-core monitoring functions.

The instrument setpoint methodologies ~~used for the US EPR were submitted to NRC in References 1 and 4. The majority of PS trips or protection functions are based on single channel inputs; therefore, the uncertainties identified in Section 3.1 of Reference 1 are applicable for the trip. Reference 4 addresses the protection system trips or protection functions that are based on multiple inputs. The uncertainty calculations for the SPNDs, incore instrumentation, high linear power density, high~~are discussed in the SCP.

BASES

BACKGROUND (continued)

~~core power level, low saturation margin, anti-dilution, and DNBR use the statistical methodology described in Reference 4. As described therein, the LTSP is the LSSS since all known errors are appropriately combined in the total loop uncertainty calculation.~~

~~LTSPs in accordance with the Allowable Value will ensure~~The SCP ensures that appropriate settings are used for Trip/Actuation Functions and that SLs of Chapter 2.0, "Safety Limits (SLs)," are not violated during AOOs, and the consequences of postulated accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or postulated accident and the equipment functions as designed.

~~Note that the Allowable Values is the least conservative value of the as-found setpoint that a Trip/Actuation Function can have during a periodic CALIBRATION or SOT, such that a Trip/Actuation Function is OPERABLE if the as-found setpoint is conservative with respect to the Allowable Value.~~

Functional testing of the entire PS, from sensor input through the opening of individual sets of Reactor Trip Circuit Breakers (RTCB) or contactors, is performed each refueling cycle. Processing transmitter CALIBRATION is also normally performed on a refueling basis.

~~Trip Setpoints that directly protect against violating the reactor core or RCS pressure boundary Safety Limits during AOOs are SL-LSSS. Permissive setpoints allow bypass of trips when they are not required by the Safety Analysis. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventative or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy, (i.e. the value indicated is sufficiently close to the necessary value to ensure proper operation of the safety systems to turn the AOO). Therefore permissives and interlocks are not considered to be SL-LSSS.~~

Manual Actuation Switches

Manual controls necessary to perform the manual operator actions credited in the safety analysis are included within the scope of the Technical Specifications. Manual actuation switches are provided to initiate the reactor trip function from the main control room (MCR) and the remote shutdown station (RSS). The ability to manually initiate ESF systems is provided in the MCR. Manual actuation of ESF systems initiates all actions performed by the corresponding automatic actuation including starting auxiliary or supporting systems and performing required sequencing functions.

Signal Processors

The PS is a distributed, redundant computer system. It consists of four independent redundant data-processing automatic paths (divisions), each with layers of operation and running asynchronous with respect to each other. In addition to the computers associated with the automatic paths, there are two redundant message and service interface computers to interface with each division.

The measurement channels or signal acquisition layer (which includes the RAUs) in each division acquires analog and binary input signals from sensors in the plant (such as for temperature, pressure, and level measurements). Each signal acquisition computer distributes its acquired and preprocessed input signals to the PS logic and controls, which includes the data processing computers (APUs).

BASES

BACKGROUND (continued)

The data-processing computers (APUs) perform signal processing for plant protective functions such as signal online validation, limit value monitoring and closed-loop control calculations. Each PS division contains four ALUs, two assigned to each subsystem. Two ALUs of the same subsystem within a division are redundant and perform the same processing using the same inputs. The outputs of two redundant ALUs are combined in a hardwired “functional AND” logic for reactor trip functions and in a hardwired OR logic for ESF functions. This avoids both unavailability of ESF functions and spurious reactor trips. The data processing computers then send their outputs to two independent voter computer units (ALUs) in each division.

In the voter computers, the outputs of the data-processing computers of redundant (three or four) divisions are processed together. A voter computer controls a set of actuators. Each voter receives the actuation signal from each of the redundant data-processing computers. The voter's task is to compare this redundant information and compute a validated (voted) actuating signal, which is used for actuating the end devices.

Each PS division contains four ALUs, two assigned to each subsystem. The two ALUs of the same subsystem within a division are redundant and perform the same processing using the same inputs. The outputs of two redundant ALUs are combined in a hardwired “functional AND” logic for reactor trip functions and in a hardwired OR logic for ESF functions.

For the reactor trip function, both ALUs in a division, if OPERABLE, must vote for an actuation. This provides protection against spurious trips. However, if only one ALU in a division is OPERABLE, the division is still OPERABLE, and the single voting ALU will initiate a reactor trip. For the ESF functions, an actuation will occur if either of the ALUs in a division votes for an actuation. This provides protection against ESF unavailability.

Reactor Trip Logic

Critical plant parameters such as temperatures, pressures, and levels are sensed, acquired, and converted to electrical signals by the PS. These signals are sent to various reactor trip functions in the PS where they are processed. When prohibited operating conditions exist, a reactor trip signal is generated from the reactor trip functions. Besides being generated automatically from the PS, a reactor trip signal can also be generated from the following systems:

BASES

BACKGROUND (continued)

- Automatic reactor trip from SAS in the event that the PS is lost;
- Manual trip from the Safety Information and Control System (SICS) panel. Four reactor trip switches are provided, which correspond to each of the four divisions;
- Manual trip from the control room; and
- Manual trip from the RSS. Note that the RSS manual trip is not part of the required circuits for LCO 3.3.1.

The reactor trip functions will utilize voting logic in order to screen out potential upstream failures of sensors or processing units. The architecture of the PS, as well as logic implemented in the system, will guard against spurious reactor trip orders while ensuring that those orders will be available when needed.

Single failures upstream of the ALU layer that could result in an invalid signal being used in the reactor trip actuation are marked as faulted by modifying the vote in the ALU layer. For the reactor trip functions, the vote is always modified toward actuation.

ESF Trip Logic

The ESF trip logic senses accident situations and initiate the operation of necessary features. The ESF along with reactor trip ensure the following:

- The integrity of the reactor coolant pressure boundary;
- The capability to shut down the reactor and maintain it in a safe shutdown condition; and
- The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures.

BASES

BACKGROUND (continued)

As with the reactor trip logic, critical plant parameters such as temperatures, pressures, and levels are sensed, acquired, and converted to electrical signals by the PS. When prohibited operating conditions exist, an ESF signal is generated from the PS. In addition to the automatic ESF actuation functions performed by the PS, the capability to manually initiate these functions is provided in the MCR. These manual functions are implemented at the system level and perform the same actions as the automatic functions. The implementation of manual system level actuation of ESF functions and the priority between the automatic functions of the PS and the manual system level initiation is determined on a case-by-case basis.

Single failures upstream of the ALU layer that could result in an invalid signal being used in the ESF actuation are marked as faulted by modifying the vote in the ALU layer. For the ESF functions, the vote is modified toward actuation except:

- The Main Steam Relief Train (MSRT) divisions, which degrade towards isolation; and
- Pressurizer Safety Relief Valve (PSRV) opening for cold overpressure protection, which degrades towards non-actuation.

<u>Actuation Devices</u>

Reactor Trip Actuation Devices

The reactor trip actuation is performed by interrupting electrical power to the Control Rod Drive Mechanisms (CRMD). Electrical power to the CRDM is delivered by the Control Rod Drive Power Supply System (CRDPSS). The CRDPSS consists of 220 V DC distribution boards which are fed from the Uninterruptible Power Supply System.

BASES

BACKGROUND (continued)

The power supply of the CDRM can be switched off via the following features:

- Four main trip breakers distributed in two electrical divisions. Two breakers are located in Division 2, two others in Division 3. The main trip breakers can be opened by two coils: one with a de-energized logic using an under voltage coil and the other with an energized logic using a shunt trip coil.

- Four trip contactors combined in a 2-out-of-4 logic feed a group of four CRDM. Division 1, 2, and 3 contains eleven groups of four CRDMs. Division 4 contains eleven groups of four CRDMs and one single CRDM for the central rod. There are a total of 92 contactors. Each trip contactor is switched off by a de-energized coil.

- The electronics of the RodPilot can switch-off the power supply of four CRDMs. Two groups of four commands can actuate this electronic module, one with low active and one with high active logic. The electronics of the RodPilot is a non-safety device of the reactor trip but is the fastest switching device and allows the contactors and the trip breaker to open without stress.

- The under voltage coil of the main trip breakers is actuated by the automatic reactor trip signals of the PS and the manual trip from the SICS panel. The shunt coil of the main trip breakers is actuated by the automatic reactor trip signal from the SAS and the manual trip signal from the RSS. The shunt coil of the trip breakers receives two different signals from SAS and RSS combined in an "OR" logic performed at the level of trip breakers.

The operator can manually close the breakers by individual controls. This control actuates the closing coil of the breaker via the SAS. In the electronics of the breaker, the opening of trip breaker must have priority to the closing.

The reactor trip signal generated automatically by the PS and the manual trip signal generated from the SICS panel can actuate the trip contactors.

Engineered Safety Features Actuation Devices

The ESF determines the need for actuation in each of the input divisions monitoring each actuation parameter. Once the need for actuation is determined, the condition is transmitted to automatic actuation output logic divisions, which perform the logic to determine the actuation of each end device. Each end device has its own automatic actuation logic.

BASES

BACKGROUND (continued)

Each of the PS sensors, signal processors, or actuation devices can be placed in lockout, which renders the component inoperable. The digital signals within the PS carry a value and a status. The signal status can be propagated through the software function blocks; therefore, if an input signal to a function block has a faulty status, the output of the function block also has a faulty status. When a signal with a faulty status reaches the voting function block, the signal is disregarded through modification of the voting logic. Individual function computers can be put into a testing and diagnostic mode via the service unit. The function processor that is being tested then behaves like a computer with a “detected fault” for the system. The signal outputs are disabled and those sent via the communication means are marked with the status “TEST” or “ERROR” and therefore masked by selection blocks with active status processing. In this case the receiving function processor behaves as if the transmitting function processor had failed.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The PS is designed to ensure that the following operational criteria are met:

– The associated actuation will occur when the parameter monitored by each division reaches its setpoint and the specific coincidence logic is satisfied; and

– Separation and redundancy are maintained to permit a division to be out of service for testing or maintenance while still maintaining redundancy within the PS instrumentation network.

Each of the analyzed transients and accidents can be detected by one or more PS Functions. Each of the PS reactor trip and ESF Functions included in the Technical Specifications are credited as part of the primary success path in the accident analysis. Non-credited functions are purely equipment protective, and their use minimizes the potential for equipment damage. Non-credited functions are not included in the Technical Specifications. Refer to FSAR Sections 7.2 and 7.3.

The LCO requires the PS sensors, manual actuation switches, signal processors, and specified actuation devices to be OPERABLE. The LCO ensures that each of the following requirements is met:

– A reactor trip or ESF function will be initiated when necessary; and

– Sufficient redundancy is maintained to permit a component to be out of service for testing or maintenance.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Failure of any sensors, signal processors, or actuation device reduces redundancy or renders the affected division(s) inoperable.

~~Trip Setpoints that directly protect against violating the reactor core or RCS pressure boundary SLs during AOOs are SL-LSSS. Permissive and interlock setpoints allow bypass of trips when they are not required by the Safety Analysis. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventative or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy, (i.e. the value indicated is sufficiently close to the necessary value to ensure proper operation of the safety systems to turn the AOO). Therefore permissives and interlocks are not considered to be SL-LSSS. Each LTSP specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip Function. The methodologies for considering uncertainties are defined in References 1 and 4.~~The Limiting Trip Setpoints, Allowable Values, and as-left and as-found tolerances, and the methodologies to calculate these values are specified in the SCP (Specification 5.5.18).

The PS sensors, manual actuation switches, signal processors, and specified actuation devices satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii) .

The PS sensors, manual actuation switches, signal processors, and specified actuation devices that support reactor trips are required to be OPERABLE in MODES 1, 2 and/or 3 because the reactor is or can be made critical in these MODES. The automatic reactor trip functions are designed to take the reactor subcritical, which maintains the SLs during AOOs and assists the ESF in providing acceptable consequences during accidents. The PS sensors, manual actuation switches, signal processors, and specified actuation devices that support automatic reactor trip functions are not required to be OPERABLE in MODES 4 and 5. In MODES 4 and 5, the emphasis is placed on return to power events. The reactor is protected in these MODES by ensuring adequate SDM.

The PS sensors, manual actuation switches, signal processors, and specified actuation devices that support reactor trips are required to be OPERABLE in MODES 1, 2, 3 and/or 4 since there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the MSIVs to preclude a positive reactivity addition,

- Actuate Emergency Feedwater (EFW) to preclude the loss of the SGs as a heat sink (in the event the normal feedwater system is not available),

- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB), and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

In MODES 5 and 6, automatic actuation of the ESF Functions is not normally required because adequate time is available to evaluate plant conditions and respond by manually operating the ESF components if required. Exceptions to this are:

- ESF 10.a - Emergency Diesel Generator (EDG) Start on Degraded Grid Voltage,

- ESF 10.b - EDG Start on Loss of Offsite Power (LOOP),

- ESF 11.b - Chemical and Volume Control System (CVCS) Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating),

- ESF 11.c - CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions,

- ESF 12.a and 12.b - PSRV Actuation - First and Second Valve, and

- ESF 13 - Control Room Heating, Ventilation and Air Conditioning (HVAC) Reconfiguration to Recirculation Mode on High Intake Activity.

These ESF functions are required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies to ensure that:

- Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

- Systems needed to mitigate a fuel handling accident are available; and

- Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The specific safety analysis and OPERABILITY requirements applicable to each PS protective function is identified below.

A. REACTOR TRIPS

1. Low DNBR (Includes High Outlet Quality)

This function protects the fuel against the risk of departure from nucleate boiling during AOOs that lead to a decrease of the DNBR value. There are five Low DNBR trips:

a. Low DNBR,
b. Low DNBR and Imbalance or Rod Drop,
c. Variable Low DNBR and Rod Drop,
d. Low DNBR - High Quality, and
e. Low DNBR - High Quality and Imbalance or Rod Drop.

Together, these five trips protect against the following AOOs:

- Increase in heat removal by the secondary system,
- Decrease in heat removal by the secondary system,
- Reactivity and power distribution anomalies, and
- Decrease in reactor coolant inventory.

The Low DNBR (1.a) and High Quality (1.d) trips require four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- SPNDs,
- RCP speed sensor,
- Pressurizer Pressure (Narrow Range) sensor,
- Cold leg temperature (Narrow Range) sensor,
- RCS loop flow sensors,
- RAU,
- APUs, and
- ALUs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low DNBR and Imbalance or Rod Drop (1.b), Variable Low DNBR and Rod Drop (1.c), and High Quality and Imbalance or Rod Drop (1.e) trips require four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- SPNDs,
- Rod Cluster Control Assembly (RCCA) position indicators,
- RCP speed sensor,
- Pressurizer Pressure (Narrow Range) sensor,
- Cold leg temperature (Narrow Range) sensor,
- RCS loop flow sensors,
- RAU,
- RCCA Unit,
- APU, and
- ALUs.

The ~~LTSPs~~ Analytical Limits are high enough to provide an operating envelope that prevents an unnecessary low DNBR reactor trip. The Analytical Limits~~LTSPs~~ are low enough for the system to maintain a margin to unacceptable fuel cladding damage for AOOs that leads to an uncontrolled decrease of the DNBR value.

The P2 permissive automatically enables the five Low DNBR Trip signals when the neutron flux, as measured by the power range, is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trips are also automatically disabled by Permissive P2.

2. High Linear Power Density

This function protects the fuel against the risk of melting at the center of the fuel pellet, during accidental transients, for events leading to an uncontrolled increase of the linear power density.

This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Reactivity and power distribution anomalies.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The High Linear Power Density Trip requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- SPNDs,
- RAU,
- APUs, and
- ALUs.

The Analytical Limits~~LTSPs~~ are high enough to provide an operating envelope that prevents unnecessary High Linear Power reactor trips. The Analytical Limits~~LTSPs~~ are low enough for the system to maintain a margin to unacceptable fuel centerline melt for any AOOs that lead to an uncontrolled increase of the linear power density.

The P2 permissive automatically enables the Reactor Trip signal when the neutron flux, as measured by the power range, is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trip is also automatically disabled by Permissive P2.

3. High Neutron Flux Rate of Change (Power Range)

This function limits the consequences of an excessive reactivity increase from an intermediate power level including nominal power. This trip protects against reactivity and power distribution anomalies.

The High Neutron Flux Rate of Change Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3 with the Reactor Control, Surveillance and Limitation (RCSL) System capable of withdrawing a RCCA or one or more RCCAs not fully inserted:

- Power Range sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary Excore High Neutron Flux Rate of Change reactor trips. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to unacceptable fuel cladding damage due to an excessive reactivity increase from an intermediate power level including nominal power.

There are no permissives associated with this trip.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. High Core Power Level

This function limits the consequences of an excessive reactivity increase from an intermediate high power level including nominal power. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Reactivity and power distribution anomalies.

The High Core Power Level Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and in MODE 2 when the nuclear power level is greater than or equal to 10^{-5}% power as indicated on the Intermediate Range monitors:

- Cold Leg Temperature sensors (Wide Range),
- Hot Leg Temperature (Narrow Range) sensors,
- Hot Leg Pressure (Wide Range) sensors,
- RCS Loop Flow sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents an unnecessary High Core Power Level reactor trip. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to unacceptable fuel cladding damage due to an excessive reactivity increase from an intermediate high power level including nominal power.

The P5 permissive automatically enables the High Core Power Level Trip when the nuclear power level is greater than or equal to 10^{-5}% power. The P5 permissive also automatically disables the High Core Power Level Trip below this power.

5. Low Saturation Margin

This function limits the consequences of an excessive reactivity increase from an intermediate high power level including nominal power. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Reactivity and power distribution anomalies.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low Saturation Margin Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODE 2 when the nuclear power level is greater than or equal to 10^{-5}% power as indicated on the Intermediate Range monitors.:

- Cold Leg Temperature sensors (Wide Range),
- Hot Leg Temperature (Narrow Range) sensors,
- Hot Leg Pressure (Wide Range) sensors,
- RCS Loop Flow sensors,
- APUs, and
- ALUs.

The ~~LTSP~~ Design Limit is high enough to provide an operating envelope that prevents an unnecessary Low Saturation Margin reactor trip. The ~~LTSP~~ Design Limit is low enough for the system to maintain a margin to unacceptable fuel cladding damage during AOOs.

The P5 permissive automatically enables the Low Saturation Margin Trip when the nuclear power level is greater than or equal to 10^{-5}%. The P5 permissive also automatically disables the Low Saturation Margin Trip below this power.

6. RCS Loop Flow Rate

This function initiates a reactor trip and is inhibited below a certain level of nuclear power under which the protection is not necessary because DNB is no longer a risk in this condition. There are two trips:

a. Low-Low RCS Loop Flow Rate in One Loop, and
b. Low RCS Loop Flow Rate in Two Loops.

These trips protect against the following postulated accidents or AOOs:

- Decrease in heat removal by the secondary system, and
- Decrease in RCS flow rate.

The Low-Low RCS Loop Flow in One Loop Trip (6.a) requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 70% RTP:

- RCS Loop Flow sensors,
- APUs, and
- ALUs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary Low-Low Loop Flow Rate reactor trips. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to ensure DNBR limits are met for AOOs and bounded for postulated accidents.

The P3 permissive automatically enables the Low-Low RCS Loop Flow Rate Trip (One Loop) when the nuclear power level is greater than or equal to 70% RTP. The P3 permissive also automatically disables the Low-Low RCS Loop Flow Rate Trip (One Loop) below this power.

The Low RCS Loop Flow in Two Loops Trip (6.b) requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- RCS Loop Flow sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary Low Loop Flow Rate reactor trips. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to ensure DNBR limits are met for AOOs.

The P2 permissive automatically enables the Low RCS Loop Flow Rate Trip (Two Loops) when the nuclear power level is greater than or equal to 10% RTP. The P2 permissive also automatically disables the Low RCS Loop Flow Rate Trip (Two Loops) when the nuclear power level is below this power.

7. Low RCP Speed

Due to electrical transients that may affect the RCP's, a specific protection function is required. This function initiates a reactor trip and is inhibited below a low level of reactor power under which the protection is not necessary because DNB is no longer a risk.

This trip protects against the following postulated accidents or AOOs:

- Decrease in heat removal by the secondary system, and
- Decrease in RCS flow rate.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low RCP Speed Trip requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- RCP Speed Trip sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary Low RCP Speed reactor trips. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to ensure DNBR limits are met for AOOs.

The P2 permissive automatically enables the Low RCP Speed Trip when the power level is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trip is also automatically disabled by permissive function P2.

8. High Neutron Flux (Intermediate Range)

This function limits the consequences of an excessive reactivity increase when the reactor is started up from a sub-critical or low power start-up condition. This trip protects against reactivity and power distribution anomalies.

The High Neutron Flux Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 when RTP is less than or equal to 10%, MODE 2, and in MODE 3 when RCSL is capable of withdrawing a RCCA or one or more RCCAs not fully inserted:

- Intermediate Range sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents an unnecessary High Neutron Flux reactor trip. The Analytical Limit ~~LTSP~~ is low enough for the system to maintain a margin to unacceptable fuel cladding damage for AOOs that leads to an uncontrolled increase of the linear power density.

The P6 permissive automatically enables the High Neutron Flux Intermediate Range reactor trip when the power level is less than or equal to 10% RTP.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Low Doubling Time (Intermediate Range)

This function limits the consequences of an excessive reactivity increase when the reactor is started up from a sub-critical or low power start-up condition. This trip protects against reactivity and power distribution anomalies.

The Low Doubling Time Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 when RTP is less than or equal to 10%, MODE 2, and in MODE 3 when RCSL is capable of withdrawing a RCCA or one or more RCCAs not fully inserted:

- Intermediate Range sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents an unnecessary Low Doubling Time reactor trip. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to unacceptable fuel cladding damage for any postulated event that leads to an uncontrolled increase of the linear power density.

The P6 permissive automatically enables the Low Doubling Time reactor trip when the power level is less than or equal to 10% RTP.

10. Low Pressurizer Pressure

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. This trip protects against a decrease in reactor coolant inventory.

The Low Pressurizer Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- Pressurizer Pressure (Narrow Range) sensors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. The Analytical Limit~~LTSP~~ is sufficiently below the full load operating value for RCS pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of an RCS depressurization.

The P2 permissive automatically enables the Low Pressurizer Pressure Trip when the power level is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trip is automatically disabled by permissive function P2.

11. High Pressurizer Pressure

In case of a RCS overpressure, a reactor trip is required in order to:

- Adapt the reactor power to the capacity of the safety systems;
- Ensure RCS integrity; and
- Avoid opening of the Pressurizer safety valves in certain primary side overpressure analyses.

This trip protects against a decrease in heat removal by the secondary system.

The High Pressurizer Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- Three Pressurizer Pressure (Narrow Range) sensors,
- Three divisions of APUs, and
- Three divisions of ALUs.

The Analytical Limit~~LTSP~~ is ~~set~~ below the nominal lift setting of the Pressurizer code safety valves, and its operation avoids the undesirable operation of these valves during normal plant operation. In the event of a complete loss of electrical load from 100% power, this setpoint ensures the reactor trip will take place, thereby limiting further heat input to the RCS and consequent pressure rise. The PSRVs may lift to prevent over pressurization of the RCS.

There are no permissives associated with this trip.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

12. High Pressurizer Level

In case of increasing Pressurizer level, a reactor trip is required in order to avoid Pressurizer over filling and to prevent the PSRVs from relieving. This trip protects against increases in reactor coolant inventory.

The High Pressurizer Level Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- Pressurizer Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is ~~set~~ below the point where the associated transient would reach the nominal lift setting of the PSRVs, and its operation avoids the undesirable operation of these valves during normal plant operation. In the event of a CVCS malfunction, this Analytical Limit~~setpoint~~ ensures a timely reactor trip will take place in order to avoid filing the pressurizer. The PSRVs may lift to prevent over pressurization of the RCS.

The P12 permissive automatically enables the High Pressurizer Level Trip when the pressure is greater than or equal to 2005 psia.

13. Low Hot Leg Pressure

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. This trip protects against a decrease in reactor coolant inventory.

The Low Hot Leg Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and in MODE 3 with the pressurizer pressure greater than or equal to 2005 psia, when the RCSL System is capable of withdrawing a RCCA, or one or more RCCAs are not fully inserted.

- Hot Leg Pressure (Wide Range) sensors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. The Analytical Limit~~LTSP~~ is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of abnormal conditions.

The P12 permissive automatically enables the Low Hot Leg Pressure Trip when the pressure is greater than or equal to 2005 psia.

14. Steam Generator Pressure Drop

In case of steam or feedwater system piping failure, the affected Steam Generator (SG) depressurizes leading to a RCS cooldown and hence a reactivity transient. A reactor trip is required in order to ensure the fuel rod integrity and to adapt the reactor power to the capacity of the safety systems. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Decrease in heat removal by the secondary system.

The SG Pressure Drop Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- SG Pressure sensors,
- APUs, and
- ALUs.

In case of steam or feedwater system piping failure, the affected SG depressurizes leading to a RCS cooldown or heatup. A reactor trip is required in order to ensure the fuel rod integrity and to adapt the reactor power to the capacity of the safety systems. The Analytical Limit~~LTSP~~ is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of a pipe break.

There are no permissives associated with this trip.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

15. Low SG Pressure

In case of steam or feedwater system piping failure, the affected SG depressurizes leading to a RCS cooldown and hence a criticality transient. For small breaks, the setpoint of the reactor trip on SG pressure drop may not be reached. Therefore, a reactor trip on low SG pressure is introduced in order to ensure fuel rod integrity and to adapt the reactor power to the capacity of safety systems. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Decrease in heat removal by the secondary system.

The Low SG Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and in MODE 3 with either the pressurizer pressure greater than or equal to 2005 psia, the RCSL System capable of withdrawing a RCCA, or one or more RCCAs not fully inserted:

- SG Pressure sensors,
- APUs, and
- ALUs.

In case of steam or feedwater system piping failure, the affected SG depressurizes leading to a RCS cooldown or heatup. For small breaks, the setpoint of the reactor trip on SG pressure drop may not be reached. Therefore, a reactor trip on low SG pressure is introduced in order to ensure fuel rod integrity and to adapt the reactor power to the capacity of safety systems. The Analytical Limit~~LTSP~~ is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of a pipe break.

The P12 permissive automatically enables the Low SG Pressure Trip when the pressure is greater than or equal to 2005 psia.

16. High SG Pressure

In case of a loss of the main heat sink, the reactor has to be tripped in order to:

- Ensure fuel rods integrity at power;
- Adapt the reactor power to the capacity if safety systems; and
- Ensure SG integrity.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This trip protects against a decrease in heat removal by the secondary system.

The High SG Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1:

- SG Pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is ~~set~~ high enough to avoid spurious operation. In case of a loss of the main heat sink, the Analytical Limit~~setpoint~~ is ~~set~~ low enough to trip the reactor in order to:

- Ensure fuel rod integrity at power,
- Adapt the reactor power to the capacity of safety systems, and
- Ensure SG integrity.

There are no permissives associated with this trip.

17. Low SG Level

This trip protects the reactor from a loss of heat sink in case of SG steam/feedwater flow mismatch. This trip protects against a decrease in heat removal by the secondary system.

The Low SG Level Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- SG Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The purpose of this trip is to protect the reactor from a loss of heat sink in case of SG steam/feedwater flow mismatch. The Analytical Limit~~LTSP~~ is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of a flow mismatch.

The P13 permissive automatically enables the Low SG Level Trip when the hot leg temperature is greater than or equal to 200ºF.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

18. High SG Level

This trip protects the turbine against an excessive humidity in case of a Main Feedwater (MFW) malfunction causing an increase in feedwater flow or in case of SG level increase. This reactor trip ensures core integrity during these transients since an increase in feedwater flow leads to a RCS overcooling event and hence a reactivity insertion. This trip protects against an increase in heat removal by the secondary system.

The High SG Level Trip requires the following sensors and processors to be OPERABLE in MODE 1 and in MODE 2:

- SG Level (Narrow Range) sensors
- APUs, and
- ALUs.

This reactor trip ensures core integrity during transients involving a MFW malfunction that results in an increase in feedwater flow or in case of a SG level increase. The Analytical Limit~~LTSP~~ is sufficiently ~~below~~ above the full load operating value so as not to interfere with normal plant operation, but still ~~high~~ low enough to provide the required protection in the event of an abnormal condition.

The P13 permissive automatically enables the High SG Level Trip when the hot leg temperature is greater than or equal to 200ºF.

19. High Containment Pressure

In case of a postulated initiating event leading to water or steam discharge into the containment, a reactor trip is performed in order to ensure containment integrity and to adapt the reactor power to the capacity of the safety systems. This trip protects against the following postulated accidents or AOOs:

- Decrease in heat removal by the secondary system, and
- Decrease in reactor coolant inventory.

The High SG Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- Containment Equipment Compartment and Containment Service Compartment pressure sensors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In case of a postulated initiating event leading to water or steam discharge into the containment, a reactor trip is performed in order to ensure containment integrity and to adapt the reactor power to the capacity of the safety systems. The Analytical Limit~~LTSP~~ is ~~set~~ high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an abnormal condition. It is set low enough to initiate a reactor trip when an abnormal condition is indicated.

There are no permissives associated with this trip.

20. Manual Reactor Trip

A manual reactor trip signal can be generated from the SICS panel and the RSS. The manual trip signal from the RSS actuates a reactor trip through energizing the shunt coils of the main reactor trip breakers.

B. ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) FUNCTIONS

Each of the analyzed accidents or AOOs can be detected by one or more ESF Functions. One of the ESF Functions is the primary actuation signal for that accident. An ESF Function may be the primary actuation signal for more than one type of accident. An ESF Function may also be the secondary, or backup, actuation signal for one or more other accidents. The ESF protective functions are described below.

1. Turbine Trip on Reactor Trip

A turbine trip is required following any reactor trip in order to avoid a mismatch between primary and secondary power, which would result in excessive RCS cooldown with a potential return to critical conditions and power excursion.

The automatic Turbine Trip on Reactor Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1:

- RTCB Position Indication sensor,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A turbine trip is required following any reactor trip in order to avoid a mismatch between primary and secondary power. Such a mismatch could result in an RCS cooldown transient, with a potential inadvertent return to critical conditions. The one second time delay is an Analytical Limit.

There are no automatic permissives associated with this function.

2. Main Feedwater

a. MFW Full Load Closure on Reactor Trip (All SGs)

After a reactor trip check-back, a MFW full load isolation is required. This avoids a mismatch between primary and secondary power. Such a mismatch could result in an RCS cooldown transient, with a potential inadvertent return to critical conditions.

The automatic MFW Full Load Closure on Reactor Trip function requires four divisions of the following processors to be OPERABLE in MODE 1 and MODE 2 except when the MFW full load isolation valves are closed:

- RTCB Position Indication sensor,
- APUs, and
- ALUs.

There are no automatic permissives associated with this function.

b. MFW Full Load Closure on High SG Level (Affected SG)

In the case of an increasing SG level event, the MFW supply to the affected SG is isolated in order to avoid filling the SG, and subsequently introducing water into Main Steam line and MSRT.

This function mitigates an increase in heat removal from the secondary system.

The automatic MFW Full Load Closure on High SG Level function requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODES 2 and 3, except when all MFW full load and low load isolation valves are closed:

- SG Level sensors,
- APUs, and
- ALUs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The MFW Full Load Closure on High SG Level Analytical Limit~~LTSP~~ is ~~set~~ high enough to avoid spurious actuation but low enough in order to prevent water level in the SG from rising and entering the steam line.

The P13 permissive automatically enables the MFW Full Load Closure on High SG Level function when the hot leg temperature is greater than or equal to 200 ºF.

c. Startup and Shutdown Feedwater Isolation on SG Pressure Drop (All SGs)

The affected SG depressurizes for the listed events, a reactor trip is initiated on a SG pressure drop signal. Also, the Startup and Shutdown Feedwater (SSS) isolation and control valves close in all the SGs.

A complete Feedwater system isolation in the affected SG limits the coolant provided into the affected SG by the MFW/SSS. This action minimizes the mass and energy released into the containment and RCS cooldown.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Steam system piping failure, and
- Feedwater system piping failure.

The automatic SSS Feedwater Isolation on SG Pressure Drop function requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODES 2 and 3, except when all MFW full load and low load isolation valves are closed:

- SG pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to preclude spurious operation but low enough to terminate feedwater flow before overcooling of the primary system or depletion of secondary inventory.

There are no automatic permissives associated with this function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. SSS Isolation on Low SG Pressure (All SGs)

The affected SG depressurizes in the event of a steam line or Feedwater pipe failure. In the event of a small secondary side break for which the SG pressure drop signal is never reached, this function also isolates the SSS supply to the affected SG. This action minimizes the mass and energy released into the containment.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Steam system piping failure, and
- Feedwater system piping failure.

The automatic SSS Feedwater Isolation on Low SG Pressure function is required to be OPERABLE in:

- – MODES 1,
- – MODE 2, except when all MFW low load isolation valves are closed, and
- – MODE 3 when the pressurizer pressure is greater than or equal to 2005 psia, except when all MFW low load isolation valves are closed.

The automatic SSS Feedwater Isolation on Low SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE:

- – SG pressure sensors,
- – APUs, and
- – ALUs.

The Analytical Limit~~LTSP~~ is high enough to preclude spurious operation but low enough to terminate feedwater flow before overcooling of the primary system or depletion of secondary inventory.

The P12 permissive automatically enables the SSS Isolation on Low SG Pressure function when the pressurizer pressure is greater than 2005 psia.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

e. SSS Isolation on High SG Level for Period of Time (Affected SGs)

During an increase in SG level after a reactor trip, the SSS systems are isolated in the affected SG in order to avoid the SG filling up and thus carryover of water into Main Steam line and subsequent water discharge by MSRT. This function mitigates Increase in Feedwater flow.

The automatic SSS Isolation on High SG Level for Period of Time function requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODES 2 and 3, except when all MFW low load isolation valves are closed:

- RTCB Position Indication,
- SG Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The SSS Isolation on High SG Level for Period of Time Analytical Limit~~LTSP~~ is ~~set~~ high enough to avoid spurious actuation but low enough in order to prevent water level in the SGs from rising and entering the steam lines.

The P13 permissive automatically enables the SSS Isolation on High SG Level for Period of Time function when the hot leg temperature is greater than 200 ºF.

3. Safety Injection System Actuation

a. Low Pressurizer Pressure

In the event of a decrease in RCS water inventory, the makeup is supplied by the Medium Head Safety Injection (MHSI) in the high pressure phase of the event and the Low Head Safety Injection (LHSI) in the low pressure phase. For a potential overcooling event, the reactivity insertion is limited by the boron injection via the MHSI. Even if the boron injection is not required, MHSI injection is needed to stabilize the RCS pressure.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Safety Injection System (SIS) Actuation function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- MSLB,
- Feedwater Line Break,
- Inadvertent opening of a pressurizer pilot operated safety valve,
- Small break LOCA,
- Steam system piping failure, and
- Large break LOCA.

The automatic SIS Actuation on Low Pressurizer Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3 with the pressurizer pressure greater than or equal to 2005 psia:

- Three Pressurizer Pressure (Narrow Range) sensors,
- Three divisions of APUs, and
- Three divisions of ALUs.

The Analytical Limit~~LTSP~~ for this function is set below the full load operating value for RCS pressure so as not to interfere with normal plant operation. However, the Analytical Limit~~setting~~ is high enough to provide an SIS actuation during an RCS depressurization.

The P12 permissive automatically enables the SIS Actuation on Low Pressurizer Pressure function when the pressurizer pressure is greater than or equal to 2005 psia.

The capability for manual initiation of the SIS is provided to the operator in the MCR. This manual initiation starts the four trains of SI. Four manual initiation controls are provided, any two of which will start the four SIS trains.

b. Low Delta P_{sat}

This function ensures SIS actuation in the hot and cold shutdown conditions with LHSI / Residual Heat Removal (RHR) in operation and at least one RCP operating.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This function mitigates the following postulated accidents or AOOs:

- Small break LOCA,
- Large break LOCA,
- Spurious opening of one Main Steam relief or safety valve,
- Inadvertent opening of a pressurizer pilot operated safety valve,
- Excessive increase in secondary steam flow, and
- MSLB.

The automatic SIS Actuation on Low Delta P_{sa} function requires four divisions of the following sensors and processors:

- Hot Leg Pressure (Wide Range) sensors,
- Hot Leg Temperature (Wide Range) sensors,
- APUs, and
- ALUs.

These sensors and processors are required to be OPERABLE in MODE 3 when Trip/Actuation Function B.3.a, SIS Actuation on Low Pressurizer Pressure, is disabled.

This function ensures SIS actuation in the hot and cold shutdown conditions with LHSI/RHR in operation and at least one of the RCPs are operating.

The Analytical Limit~~LTSP~~ for the Low Delta P_{sat} function is ~~set~~ high enough to avoid spurious operation but low enough to maintain core coverage in the event of an RCS pipe break.

The P12 permissive automatically enables the SIS Actuation on Low Delta P_{sat} function when the pressurizer pressure is less than or equal to 2005 psia. The P15 permissive automatically enables the SIS Actuation on Low Delta P_{sat} function when at least two RCPs are running, the hot leg pressure is greater than or equal to 464 psia, and when the hot leg temperature is greater than or equal to 356ºF.

The capability for manual initiation of the SIS is provided to the operator in the MCR. This manual initiation starts the four trains of SI. Four manual initiation controls are provided, any two of which will start the four SIS trains.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. RCP Trip on Low Delta-Pressure across the RCP with SIS Actuation

In case of LOCA in combination with a SIS actuation, the RCPs are tripped to prevent their operation in scenarios where timing of the pump trip is related to maintaining core cooling.

This function mitigates the following postulated accidents or AOOs:

- Inadvertent opening of a PSRV, and
- Small break LOCA.

The automatic RCP Trip on Low Delta-Pressure across RCP with SIS Actuation function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- RCP Delta-Pressure sensors,
- RCP Current sensors,
- APUs, and
- ALUs.

The Analytical Limit ~~LTSP~~ for the RCP Trip on Low Delta-Pressure across RCP with SIS Actuation function is ~~set~~ high enough to avoid spurious operation but low enough to ensure core cooling is maintained.

There are no automatic permissives associated with this function.

5. Partial Cooldown on SIS Actuation

The partial cooldown consists of lowering the MSRT setpoint down to allow depressurization of the RCS by heat removal of the SGs. This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- MSLB,
- Inadvertent opening of a Pressurizer pilot operated safety valve, and
- Small break LOCA.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic Partial Cooldown on SIS Actuation function requires four divisions of the following processors to be OPERABLE in MODES 1, 2, and 3:

- APUs, and
- ALUs.

The LTSP for the Partial Cooldown Actuation on SIS Actuation function is set high enough to avoid spurious operation but low enough to ensure adequate flow from the MHSI pumps to maintain core cooling.

The P14 permissive automatically enables the Partial Cooldown on SIS Actuation function when the hot leg pressure is greater than or equal to 464 psia and the hot leg temperature is greater than or equal to 356 ºF.

6. Emergency Feedwater System

a. Actuation on Low-Low SG Level (All SGs)

In case of loss of MFW, the Emergency Feedwater System (EFWS) is actuated to remove residual heat via secondary side. With an EFWS actuation signal, SG blowdown is also isolated to conserve SG inventory. This function mitigates the following postulated accidents or AOOs:

- Loss of normal feedwater flow,
- Feedwater system piping failure, and
- LOOP.

The automatic EFWS Actuation on Low-Low SG Level function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2 and 3:

- SG level (Wide Range) sensors,
- APUs, and
- ALUs.

This function ensures heat is removed from the primary system through the SGs in the event of a loss of MFW or feedwater line break, as indicated by low SG level. The Analytical Limit ~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary actuations but low enough to ensure sufficient make-up is provided to the SGs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The P13 permissive automatically enables the EFWS Actuation on Low-Low SG Level function when the hot leg temperature is greater than or equal to 200ºF.

b. Actuation on LOOP and SIS Actuation (All SGs)

The LOOP results in a trip of the turbine, RCPs, and MFW pumps. The MFW and SSS supply cut off leads to a decrease in secondary side heat removal and the primary flow coast down further reduces the capacity of the primary coolant to remove heat from the core. As a result, primary and secondary pressures and temperatures increase. The heat is removed via MSRT and EFWS. With an EFWS actuation signal, SG blowdown is also isolated to conserve SG inventory.

This function mitigates the consequences of a Small Break LOCA.

The automatic EFWS Actuation on LOOP and SIS function requires four divisions of the following processors to be OPERABLE in MODES 1 and 2:

- 6.9 kV Bus Voltage sensors,
- APUs, and
- ALUs.

This function ensures heat is removed from the primary system through the SGs in the event of a LOCA concurrent with a LOOP.

There are no automatic permissives associated with this function.

c. Isolation on High SG Level (Affected SG)

In the case of an increasing SG level event, the EFWS supply to the affected SG is isolated in order to avoid filling the SG, and subsequently introducing water into Main Steam line and MSRT. This function precludes overfilling of the SG.

The automatic EFWS Isolation on High SG Level function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2 and 3:

- SG level (Wide Range) sensors,
- APUs, and
- ALUs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This function ensures the SGs are not overfilled, which could allow radioactive water to be discharged through the MSRTs. The ~~LTSP~~ Design Limit is high enough to provide an operating envelope that prevents unnecessary isolations but low enough to ensure the SGs are not over-filled.

The P13 permissive automatically enables the EFWS Isolation on High SG Level function when the hot leg temperature is greater than or equal to 200 ºF.

7. Main Steam Relief Train

a. Actuation on High SG Pressure

In the event of a loss of the secondary side heat sink, the residual heat is removed through the steam relief valves to the atmosphere. This is done by the MSRT. The MSRT also ensures SG overpressure protection, minimizes the actuation of the Main Steam Safety Valves (MSSVs), which reduces the risk of a stuck open safety relief valve.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Total loss of load and/or turbine trip
- Loss of main heat sink (condenser),
- Inadvertent closure of a Main Steam Isolation Valve (MSIV),
- MSLB,
- RCP seizure (locked rotor) or RCP shaft break., and
- Feedwater system piping failure.

The automatic MSRT Actuation on High SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- SG Pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ for the MSRT Actuation on High SG Pressure function is set high enough to avoid spurious operation and low enough to open and relieve SG pressure before over pressurization limits are reached.

There are no automatic permissives associated with this function.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Isolation on Low SG Pressure

The Main Steam Relief Isolation Valves (MSRIVs) are opened during events in order to control pressure in the SGs. In order to prevent a stuck open Main Steam Relief Control Valve from causing an RCS cooldown and a risk of return to critical conditions, the MSRT is isolated.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Loss of main heat sink (condenser),
- Inadvertent Opening of SG Safety or Relief Valve, and
- MSLB.

The automatic MSRT Isolation on Low SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3 with the pressurizer pressure is greater than or equal to 2005 psia:

- SG pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ for the MSRT Isolation on Low SG Pressure function is set low enough to avoid spurious operation and high enough to limit the rate of RCS cooldown.

The P12 permissive automatically enables the MSRT Isolation on Low SG Pressure function when the pressure is greater than or equal to 2005 psia.

8. MSIV Closure

a. Closure on SG Pressure Drop (All SGs)

In case of a secondary side Steam Line or Feedwater system pipe break, the affected SG depressurizes. This function isolates all four SGs in order to:

- Prevent draining of unaffected SG,
- Limit return to criticality conditions due to a overcooling transient,
- Limit the release of radioactivity, and
- Limit mass and energy releases into the containment.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Spurious opening of one SG safety or relief valve,
- Steam system piping failure, and
- Feedwater system piping failure.

The automatic MSIV Closure on SG Pressure Drop function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- SG Pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ for the MSIV Closure on SG Pressure Drop function is ~~set~~ high enough to avoid SG pressure fluctuations during normal operation and low enough to isolate a SG and limit the blowdown to the value assumed in the safety analysis.

There are no automatic permissives associated with this function.

b. Closure on Low SG Pressure (All SGs)

For most Main Steam Line or Feedwater pipe breaks, the affected SG depressurizes. For small breaks, the setpoint for MSIV closure on SG pressure drop may not be reached. This function isolates all four SG on the main steam side in the event of a secondary side break in order to:

- Prevent draining of unaffected SGs,
- Limit the return to critical conditions due to a overcooling transient,
- Limit the release of radioactivity, and
- Limit mass and energy releases into the containment.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Spurious opening of one SG safety or relief valve,
- Steam system piping failure, and
- Feedwater system piping failure.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic MSIV Closure on Low SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3, except when all MSIVs are closed:

- SG pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ for the MSIV Closure on Low SG Pressure function is ~~set~~ high enough to avoid SG pressure fluctuations during normal operation and low enough to isolate a SG and limit the blowdown to the value assumed in the safety analysis.

The P12 permissive automatically enables the MSIV Closure on Low SG Pressure function when the pressurizer pressure is greater than or equal to 2005 psia.

9. Containment Isolation

a. Isolation (Stage 1) on High Containment Pressure

In case of a LOCA, the containment has to be isolated in order to prevent release of radioactivity to the environment. Safeguards Building HVAC is also reconfigured to process air through High Efficiency Particulate Air (HEPA) filters to ensure 10 CFR 50.34 and 10 CFR 100.21 limits are not exceeded.

The automatic Stage 1 Containment Isolation on High Containment Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- Containment Service Compartment Pressure monitors,
- Containment Equipment Compartment Pressure monitors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ for the Stage 1 Containment Isolation on High Containment Pressure function is ~~set~~ high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 and 10 CFR 100.21 limits.

There are no automatic permissives associated with this function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Isolation (Stage 1) on SIS Actuation

In case of the listed events, the containment has to be isolated in order to prevent release of radioactivity to the environment. Safeguards Building HVAC is also reconfigured to process air through HEPA filters to ensure 10 CFR 50.34 and 10 CFR 100.21 limits are not exceeded.

This function mitigates the following postulated accidents or AOOs:

- Inadvertent opening of a pressurizer pilot operated safety valve, and
- LOCA.

The automatic Stage 1 Containment Isolation on SIS Actuation function requires four divisions of the following processors to be OPERABLE in MODES 1, 2, 3, and 4:

- APUs, and
- ALUs.

There are no automatic permissives associated with this function.

c. Isolation (Stage 2) on High-High Containment Pressure

In case of a LOCA, the containment has to be isolated in order to prevent release of radioactivity to the environment.

This function mitigates the following postulated accidents or AOOs:

- Inadvertent opening of a pressurizer pilot operated safety valve, and
- LOCA.

The automatic Stage 2 Containment Isolation on High-High Containment Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- Containment Service Compartment Pressure monitors,
- Containment Equipment Compartment Pressure monitors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The ~~LTSP~~ Design Limit for the Stage 2 Containment Isolation on High-High Containment Pressure function is ~~set~~ high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 and 10 CFR 100.21 limits.

There are no automatic permissives associated with this function.

d. Isolation (Stage 1) on High Containment Radiation

In case of a significant release of radioactivity into the containment, the containment is isolated to ensure 10 CFR 50.34 and 10 CFR 100.21 limits are not exceeded.

This function mitigates the following postulated accidents or AOOs:

- Rod ejections,
- LOCA,
- MSLB inside containment, and
- Feedwater line break inside containment.

The automatic Stage 1 Containment Isolation on High Containment Radiation function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, 3, and 4:

- Containment High Range radiation monitors,
- APUs, and
- ALUs.

The ~~LTSP~~ Design Limit for the Stage 1 Containment Isolation on High Containment Radiation function is ~~set~~ high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 and 10 CFR 100.21 limits.

There are no automatic permissives associated with this function.

10. Emergency Diesel Generator

a. Start on Degraded Grid Voltage

Following the detection of degraded voltage for a period of time on one 6.9 kV bus, the EDG associated with that bus is automatically started. This function mitigates a LOOP, which is assumed to occur independently or concurrently with postulated accidents and AOOs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic EDG Start on Degraded Grid Voltage requires four divisions of the following processors to be OPERABLE in MODES 1, 2, 3, and 4 or when the associated EDG is required to be OPERABLE in accordance with LCO 3.8.2, “AC Sources - Shutdown”:

- 6.9 kV voltage sensors,
- APUs, and
- ALUs.

This function ensures AC Power is available to mitigate a postulated concurrent design basis event.

The ~~LTSP~~ Design Limit for the EDG Start on Degraded Grid Voltage is ~~set~~ high enough to avoid spurious operation but low enough to ensure that power is provided to ESF functions in the time-frame assumed in the accident analyses.

There are no automatic permissives associated with this function.

b. Start on LOOP

Following a LOOP on one 6.9 kV bus, the EDG associated with that bus is automatically started. This function mitigates a LOOP, which is assumed to occur independently or concurrently with postulated accidents and AOOs.

The automatic EDG Start on LOOP requires four divisions of the following processors to be OPERABLE in MODES 1, 2, 3, and 4 or when the associated EDG is required to be OPERABLE in accordance with LCO 3.8.2, “AC Sources - Shutdown”:

- 6.9 kV voltage sensors,
- APUs, and
- ALUs.

This function ensures AC Power is available to mitigate a postulated concurrent design basis event.

The ~~LTSP~~ Design Limit for the EDG Start on LOOP is ~~set~~ high enough to avoid spurious operation but low enough to ensure that power is provided to ESF functions in the time-frame assumed in the accident analyses.

There are no automatic permissives associated with this function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Chemical and Volume Control System Charging Line Isolation

a. Isolation on High-High Pressurizer Level

The isolation of the CVCS Charging Line on High-High Pressurizer Level is required to avoid filling of the pressurizer and subsequent water overflow through the safety valves.

This function protects against a CVCS malfunction that causes an increase in RCS water inventory.

The automatic CVCS Charging Line Isolation on High-High Pressurizer Level function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- Pressurizer Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is low enough to initiate appropriate mitigative actions in time to prevent the pressurizer from overfilling during the CVCS Malfunction event that may increases RCS inventory, but high enough to prevent spurious operations.

The P17 permissive automatically disables the CVCS Charging Line Isolation on High-High Pressurizer Level function when the cold leg temperature is less than or equal to 248 ºF.

b. Isolation on ADM - Shutdown Condition (RCP not operating)

The ADM function in the Shutdown Condition mitigates a dilution event where no RCPs are in operation. This function ensures that:

- The dilution is stopped when the protection is actuated, and
- The core remains sub-critical.

The automatic CVCS Charging Line Isolation on ADM - Shutdown Condition (RCP not operating) function is required to be OPERABLE in:

- MODES 5, with two or less RCPs in operation, and
- MODES 6.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic CVCS Charging Line Isolation on ADM - Shutdown Condition (RCP not operating) function requires the following sensors and processors:

- Boron Concentration - CVCS Charging Line sensors (4 divisions),
- Boron Temperature - CVCS Charging Line sensors (4 divisions),
- APUs (4 divisions), and
- ALUs (Divisions 1 and 4).

The Analytical Limit ~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary isolations but low enough to mitigate a dilution event in the shutdown condition where the RCPs are not in operation.

This function is required to be accompanied by Permissive P7, which represents a RCP speed shutdown condition, or an ATWS signal.

c. Isolation on ADM - Standard Shutdown Conditions

This function mitigates a homogeneous dilution event in the standard shutdown states where the RCPs are in operation. This function ensures that:

- The dilution is stopped when the protection is actuated, and
- The core remains sub-critical.

The automatic CVCS Charging Line Isolation on ADM - Standard Shutdown Conditions function is required to be OPERABLE in:

- MODES 3, with thee or more RCPs in operation,
- MODES 4, with thee or more RCPs in operation, and
- MODES 5, with thee or more RCPs in operation.

The automatic CVCS Charging Line Isolation on ADM - Standard Shutdown Conditions function requires the following sensors and processors:

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Boron Concentration - CVCS Charging Line sensors (4 divisions),
- Boron Temperature - CVCS Charging Line sensors (4 divisions),
- CVCS Charging Line Flow sensors (4 divisions),
- Cold Leg Temperature (Wide Range) sensors (4 divisions),
- APUs (4 divisions), and
- ALUs (Divisions 1 and 4).

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary isolations but low enough to mitigate a dilution event in the shutdown condition where the RCPs are in operation.

This function is required to be accompanied by a permissive signal, P8, which represents a reactor shutdown condition as indicated by RCCA position indication and disabled by the Permissive P7, which represents a RCP shutdown condition.

12.a and 12.b. PSRV Actuation - First and Second Valve

The integrity of the reactor pressure vessel must be ensured under all plant conditions. At low coolant temperature, the cylindrical part of the vessel could fail by brittle fracture before the design pressure of the RCS is reached. Therefore the low-temperature overpressure protection (LTOP) is ensured by opening of the PSRVs.

This function mitigates a low temperature overpressure event.

The automatic PSRVs Actuation function requires four divisions of the following processors to be OPERABLE when the PSRVs are required to be OPERABLE by LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)"

- Hot Leg Pressure (Wide Range) sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~s for the PSRV Actuation function are high enough to prevent spurious operation but low enough to prevent RCS over pressurization.

The P17 permissive automatically enables the PSRV Actuation function when the cold leg temperature is less than or equal to 248º F.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

13. Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity

In case of a significant release of radioactivity, the Control Room HVAC is reconfigured to ensure 10 CFR 50.34 limits are not exceeded.

This function mitigates the following postulated accidents or AOOs:

- Rod ejections,
- LOCA, and
- Line breaks outside containment.

The automatic Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, 3, 4, 5, 6, and during the movement of irradiated fuel assemblies:

- Control Room HVAC Intake Activity radiation monitors,
- APUs, and
- ALUs.

The ~~LTSP~~ Design Limit for the Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity function is ~~set~~ high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 limits.

There are no automatic permissives associated with this function.

C. PROTECTION SYSTEM PERMISSIVES

Protection System permissives are provided to ensure reactor trips and ESF are in the correct configuration for the current unit status. They back up operator actions to ensure Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the permissive Functions do not need to be OPERABLE when the associated reactor trip or ESF functions are outside the applicable MODES. The automatic permissives are:

1. P2 - Flux (Power Range) Measurement Higher than First Threshold

The P2 permissive is representative of PRD neutron flux measurements higher than a low-power setpoint value. The P2 setpoint value corresponds to the value below which transients do not lead to risk of DNB (10% RTP).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The P2 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip 1.a - Low DNBR,
- Reactor Trip 1.b - Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c - Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d - Low DNBR - High Quality,
- Reactor Trip 1.e - Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 2 - High Linear Power Density,
- Reactor Trip 6.b - Low RCS Flow Rate in Two Loops,
- Reactor Trip 7 - Low RCP Speed, and
- Reactor Trip 10 - Low Pressurizer Pressure.

To generate the permissive, neutron flux measurements from the PRDs are compared to the setpoint. When two out of four measurements are greater than the setpoint, the permissive is validated. Otherwise, it is inhibited.

The value of the permissive was selected such that AOOs do not challenge the DNBR or centerline melt limits when they occur at a core power level below the permissive value.

2. P3 - Flux (Power Range) Measurement Higher than Second Threshold

The P3 permissive is representative of PRD neutron flux measurements higher than an intermediate power setpoint value. The P3 setpoint value corresponds to the value below which loss of one reactor coolant pump does not lead to risk of DNB (70% Nuclear Power).

The P3 permissive is utilized in Reactor Trip 6.a - Low-Low RCS Flow Rate in One Loop.

To generate the permissive, neutron flux measurements from the PRDs are compared to the setpoint. When two out of four measurements are greater than the setpoint, the permissive is validated.

The value of the permissive was selected such that AOOs and postulated accidents that consider a loss of one RCP do not challenge the DNBR limit when they occur at a core power level below the permissive value (70% RTP).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. P5 - Flux (Intermediate Range) Measurement Higher than Threshold

The P5 permissive is representative of intermediate range detector (IRD) neutron flux measurements above a low-power setpoint value. The P5 setpoint value corresponds to the boundary between the operating ranges of the source range detectors and intermediate range detectors (greater than or equal to 10^{-5}% power as shown on the IRDs).

The P5 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip 4 - High Core Power Level, and
- Reactor Trip 5 - Low Saturation Margin.

To generate the permissive, neutron flux measurements from the IRDs are compared to the setpoint. When two out of four of the measurements are greater than the setpoint, the permissive is validated.

The value of the permissive defines the boundary between the operating range of the source range detectors and the operating range of the intermediate range detectors.

4. P6 - Thermal Core Power Higher than Threshold

The P6 permissive is representative of core thermal power above a low-power setpoint value corresponding to the boundary between the operating ranges of the IRDs and the PRDs (10% RTP).

The P6 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip 8 - High Neutron Flux (Intermediate Range), and
- Reactor Trip 9 - Low Doubling Time (Intermediate Range).

Hot leg pressure measurements, hot leg temperature measurements, and cold leg temperature measurements are used to calculate core thermal power. These calculated core thermal power levels are compared to the setpoint. When three out of four of the calculated core thermal power levels are greater than the setpoint, the operator is prompted to manually validate the permissive.

The value of the permissive was selected at the boundary between the operating range of the intermediate range detectors and the power range detectors.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. P7 - RCP Speed Lower than Threshold

The P7 permissive defines when reactor coolant pumps (RCPs) are no longer in operation. The P7 permissive is utilized in the following reactor trips or ESF functions:

- ESF 11.b - CVCS Charging Line Isolation on ADM at Shutdown Condition (RCP not operating), and
- ESF 11.c - CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

The RCP speed measurements (one per RCP) are compared to a setpoint (91% nominal speed). When two out of four of the measurements are less than the setpoint, the permissive is validated (i.e., indicates that two or more RCPs are turned off).

The value of the permissive was selected to establish the requirements for anti-dilution mitigation in a timely manner.

6. P8 - Shutdown RCCA Position Lower than Threshold

The P8 permissive defines the shutdown state with all rods in (ARI). The P8 permissive is utilized in ESF 11.c - CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

RCCA Bottom Position Indicator sensors are acquired in four different electrical divisions. For each division, when all rods in the shutdown banks reach the lower end position, a signal is generated. When two out of four of divisions indicate all rods in, the permissive is validated.

The P8 Permissive is characteristic of a shutdown state with ARI. With an ARI condition, this permissive enables the Anti-dilution in Standard Shutdown States function and inhibits the Anti-dilution in Power Condition" function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. P12 - Pressurizer Pressure Lower than Threshold

The P12 permissive defines the transition from hot shutdown to cold shutdown with respect to RCS pressure. The P12 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip 12 - High Pressurizer Level,
- Reactor Trip 13 - Low Hot Leg Pressure,
- Reactor Trip 15 - Low SG Pressure Trip,
- ESF 2.d - SSS Isolation on Low SG Pressure (All SGs),
- ESF 3.a - SIS Actuation on Low Pressurizer Pressure,
- ESF 3.b - SIS Actuation on Low Delta Psat,
- ESF 7.b - MSRT Isolation on Low SG Pressure
- ESF 8.b - MSIV Closure on Low SG Pressure (All SGs), and
- ESF 9.b - Containment Isolation (Stage 1) on SIS Actuation.

Pressurizer pressure measurements are compared to the P12 setpoint (2005 psia). The low SG pressure and low hot leg pressure reactor trip functions are automatically activated when the pressurizer pressure rises above the P12 permissive value.

The Permissive P12 reflects the transition from hot shutdown to cold shutdown. P12 ensures cooling by Main Steam Bypass or MSRT down to the LHSI/RHR connection temperature and to be able to depressurize the reactor coolant system to LHSI/RHR connection pressure without actuation of SIS.

8. P13 - Hot Leg Temperature Lower than Threshold

The P13 permissive defines when steam generator draining and filling operations are allowed. The P13 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip17 - Low SG Level,
- Reactor Trip 18 - High SG Level,
- ESF 2.b - MFW Full Load Closure on High SG Level (Affected SGs)
- ESF 2.e - SSS Isolation on High SG Level for Period of Time (Affected SGs),
- ESF 6.a - EFWS Actuation on Low-Low SG Level (All SGs), and
- ESF 6.c - EFWS Isolation on High SG Level (Affected SG).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Hot leg temperature (WR) measurements are compared to the P13 setpoint (200ºF).

The value of the permissive was selected in order to permit draining and filling operations during shutdown and LHSI/RHR in operation without generating protection signals.

9. P14 - Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds

The P14 permissive defines when the residual heat removal system is allowed to be connected to the RCS. The P14 permissive is utilized in ESF 5 - Partial Cooldown Actuation on SIS Actuation.

At pressures and temperatures below the setting of the P14 permissive (464 psia and 356 ºF), operation of the LHSI/RHR system is allowed.

This permissive is manually controlled.

10. P15 - RCPs Shutdown and P14

The P15 permissive defines when SI actuation due to delta Psat is disabled and SI actuation due to low loop level is enabled.

The P15 permissive is utilized in the following reactor trips or ESF functions:

- ESF 3.b - SIS Actuation on Low Delta Psat, and
- ESF 9.b - Containment Isolation (Stage 1) on SIS Actuation.

The value for Permissive P15 (50% no load current and P14 is true) represents the threshold for switching from the SIS Actuation on Low Delta P_{sat} protection to protection via the SIS Actuation on Low RCS Loop Level.

11. P17 - Cold Leg Temperature Lower than Threshold

The P17 permissive corresponds to the temperature conditions where brittle fracture protection is required. The P17 permissive is utilized in the following reactor trips or ESF functions:

- ESF 12.a - PSRV Actuation - First Valve, and
- ESF 12.b - PSRV Actuation - Second Valve.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The value for Permissive P17 is the threshold for activation of cold overpressure mitigation systems.

D. SENSORS, MANUAL ACTUATION SWITCHES, SIGNAL PROCESSORS, AND ACTUATION DEVICES

The relationship between sensors, manual actuation switches, signal processors, and actuation devices is provided below:

SENSORS

1. <u>6.9 kV Bus Voltage</u>

Three 6.9 kV Bus Voltage sensors per EDG are required to be OPERABLE in MODES 1, 2, 3, 4, and when the associated EDG is required to be OPERABLE by LCO 3.8.2. These sensors support the following functions:

- ESF 6.b: EFWS Actuation on LOOP and SIS Actuation (All SGs),
- ESF 10.a: EDG Start on Degraded Grid Voltage, and
- ESF 10.b: EDG Start on LOOP.

2. <u>Boron Concentration - CVCS Charging Line</u>

Four Boron Concentration - CVCS Charging Line sensors are required to be OPERABLE in MODES 3 and 4 with three or more RCPs in operation and in MODES 5 and 6. These sensors support the following functions:

- ESF 11.b: CVCS Charging Line Isolation on ADM at Shutdown Condition (RCP not operating), and
- ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Boron Temperature - CVCS Charging Line

Four Boron Temperature - CVCS Charging Line sensors are required to be OPERABLE in MODES 3 and 4 with three or more RCPs in operation and in MODES 5 and 6. These sensors support the following functions:

- ESF 11.b: CVCS Charging Line Isolation on ADM at Shutdown Condition (RCP not operating), and
- ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

4. CVCS Charging Line Flow

Four CVCS Charging Line Flow sensors are required to be OPERABLE in MODES 3, 4, and 5 when three or more RCPs are in operation. These sensors support ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

5. Cold Leg Temperature (Narrow Range)

Four Cold Leg Temperature (Narrow Range) sensors are required to be OPERABLE when RTP is greater than or equal to 10%. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop, and
- Permissive P6: Thermal Core Power Higher than Threshold.

6. Cold Leg Temperature (Wide Range)

Four Cold Leg Temperature (Wide Range) sensors are required to be OPERABLE in:

- MODE 1,
- MODE 2, when power is greater than or equal to $10^{-5}\%$ as shown on the intermediate range detectors, and in
- MODES 3, 4, 5, and 6 with three or more RCPs in operation.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

These sensors support the following functions and Permissives:

- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin,
- ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions, and
- Permissive P17: Cold Leg Temperature Lower than Threshold.

7. Containment Pressure

Four Containment Equipment Compartment Containment and Service Compartment Pressure sensors per area are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- Reactor Trip 19: High Containment Pressure,
- ESF 9.a: Containment Isolation (Stage 1) on High Containment Pressure, and
- ESF 9.c: Containment Isolation (Stage 2) on High-High Containment Pressure.

8. Hot Leg Pressure (Wide Range)

Four Hot Leg Pressure (Wide Range) sensors are required to be OPERABLE in Modes 1, 2, and 3, and when the PSRVs are required to be OPERABLE per LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)." These sensors support the following functions and Permissives:

- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin,
- Reactor Trip 13: Low Hot Leg Pressure,
- ESF 3.b: SIS Actuation on Low Delta P_{sat},
- ESF 12.a: PSRV Actuation - First Valve,
- ESF 12.b: PSRV Actuation - Second Valve,
- Permissive P6: Thermal Core Power Higher than Threshold,
- Permissive P14: Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds, and
- Permissive P15: RCPs Shutdown and P14.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Hot Leg Temperature (Narrow Range)

Four Hot Leg Temperature (Narrow Range) sensors in each of four divisions (16 total) are required to be OPERABLE in MODE 1 and MODE 2 when power is greater than or equal to 10^{-5}% as shown on the intermediate range detectors. These sensors support the following functions and Permissives:

- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin, and
- Permissive P6: Thermal Core Power Higher than Threshold.

10. Hot Leg Temperature (Wide Range)

Four Hot Leg Temperature (Wide Range) sensors are required to be OPERABLE in MODE 3 when Trip/Actuation Function B.3.a, SIS Actuation on Low Pressurizer Pressure, is disabled. These sensors support the following functions and Permissives:

- ESF 3.b: SIS Actuation on Low Delta P_{sat},
- Permissive P13: Hot Leg Temperature Lower than Threshold,
- Permissive P14: Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds, and

11. Intermediate Range

Four Intermediate Range sensors are required to be OPERABLE in:

- MODE 1, when power is less than or equal to 10% RTP,
- MODE 2, and in
- MODES 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

These sensors support the following functions and Permissives:

- Reactor Trip 8: High Neutron Flux (Intermediate Range),
- Reactor Trip 9: Low Doubling Time (Intermediate Range), and
- Permissive P5: Flux (Intermediate Range) Measurement Higher than Threshold.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

12. Power Range

Two Power Range sensors per division (8 total) are required to be OPERABLE in MODES 1 and 2, and in MODE 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These sensors support the following functions and Permissives:

- Reactor Trip 3: High Neutron Flux Rate of Change,
- Permissive P2: Flux (Power Range) Measurement Higher than First Threshold, and
- Permissive P3: Flux Measurement (Power Range) Higher than Second Threshold.

13. Pressurizer Level (Narrow Range)

Four Pressurizer Level (Narrow Range) sensors are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- Reactor Trip 12: High Pressurizer Level, and
- ESF 11.a: CVCS Charging Line Isolation on High-High Pressurizer Level.

14. Pressurizer Pressure (Narrow Range)

Four Pressurizer Pressure (Narrow Range) sensors are required to be OPERABLE in MODES 1 and 2 and MODE 3 when the pressurizer pressure is less than or equal to 2005 psia. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 10: Low Pressurizer Pressure,
- Reactor Trip 11: High Pressurizer Pressure,
- ESF 3.a: SIS Actuation on Low Pressurizer Pressure, and
- Permissive P12: Pressurizer Pressure Lower than Threshold.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

15. Radiation Monitor - Containment High Range

Four Containment High Range Radiation Monitors are required to be OPERABLE in MODES 1, 2, 3, and 4. These sensors support ESF 9.d: Containment Isolation (Stage 1) on High Containment Radiation.

16. Radiation Monitor - Control Room HVAC Intake Activity

Four Control Room HVAC Intake Activity Radiation Monitors are required to be OPERABLE in MODES 1, 2, 3, 4, 5, 6, and during the movement of irradiated fuel assemblies. The monitors are not required to be OPERABLE when the associated train is in the recirculation mode. These sensors support ESF 13: Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity.

17. RCP Current

Three RCP Current sensors per RCP (12 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions and Permissives:

- ESF 4: RCP Trip on Low Delta P across RCP with SIS Actuation, and
- Permissive P15: Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds and Reactor Coolant Pumps Shutdown.

18. RCP Delta P Sensors

Two RCP Delta-Pressure sensors per pump (8 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support ESF 4: RCP Trip on Low Delta P across RCP with SIS Actuation.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

19. RCP Speed

Four RCP Speed sensors are required to be OPERABLE when RTP is greater than or equal to 10%. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 7: Low RCP Speed, and
- Permissive P7: RCP Speed Lower than Threshold.

20. RCS Loop Flow

Four RCS Loop Flow sensors per loop (16 total) are required to be OPERABLE in MODE 1 and in MODE 2 when power is greater than or equal to 10^{-5}% as shown on the intermediate range detectors. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin,
- Reactor Trip 6a: Low-Low RCS Loop Flow Rate in One Loop,
- Reactor Trip 6b: Low RCS Loop Flow Rate in Two Loops, and
- Permissive P6: Thermal Core Power Higher than Threshold.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

21. RTCB Position Indication

Four RTCB Position Indication sensors are required to be OPERABLE in MODE 1 and in MODES 2 and 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These sensors support the following functions:

- ESF 1: Turbine Trip on Reactor Trip,
- ESF 2.a: MFW Full Load Closure on Reactor Trip (All SGs), and
- ESF 2.e: MFW and SSS Isolation on High SG Level for Period of Time (Affected SGs).

22. Self-Powered Neutron Detectors

Seventy two SPNDs are required to be OPERABLE when RTP is greater than or equal to 10%. These sensors support the following functions:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop, and
- Reactor Trip 2: High Linear Power Density.

23. SG Level (Narrow Range)

Four SG Level (Narrow Range) sensors per SG (16 total) are required to be OPERABLE in MODE 1 and in MODES 2 and 3, except when all MFW isolation valves are closed. These sensors support the following functions:

- Reactor Trip 17: Low SG Level,
- Reactor Trip 18: High SG Level,
- ESF 2.b: MFW Full Load Closure on High SG Level (Affected SGs), and
- ESF 2.e: SSS Isolation on High SG Level for Period of Time (Affected SGs).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

24. SG Level (Wide Range)

Four SG Level (Wide Range) sensors per SG (16 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- ESF 6.a: EFWS Actuation on Low-Low SG Level (All SGs), and
- ESF 6.c: EFWS Isolation on High SG Level (Affected SG).

25. SG Pressure

Four SG Pressure sensors per SG (16 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- Reactor Trip 14: SG Pressure Drop,
- Reactor Trip 15: Low SG Pressure,
- Reactor Trip 16: High SG Pressure,
- ESF 2.c: SSS Isolation on SG Pressure Drop (All SGs),
- ESF 2.d: SSS Isolation on Low SG Pressure (All SGs),
- ESF 7.a: MSRT Actuation on High SG Pressure,
- ESF 7.b: MSRT Isolation on Low SG Pressure,
- ESF 8.a: MSIV Closure on SG Pressure Drop (All SGs), and
- ESF 8.b: MSIV Closure on Low SG Pressure (All SGs).

MANUAL ACTUATION SWITCHES

1. Reactor Trip

Four manual Reactor Trip switches are required to be OPERABLE in MODES 1 and 2 and in MODES 3, 4, and 5 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These sensors support all reactor trip functions.

2. SIS Actuation

Four manual SIS Actuation switches are required to be OPERABLE in MODES 1, 2, 3, and 4. These sensors support the following functions:

- ESF 3.a: SIS Actuation on Low Pressurizer Pressure,
- ESF 3.b: SIS Actuation on Low Delta P_{sat}.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. SG Isolation

Four manual SG Isolation switches per SG (16 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- ESF 2.b: MFW Full Load Closure on High SG Level (Affected SGs);
- ESF 2.c. SSS Isolation on SG Pressure Drop (All SGs);
- ESF 5: Partial Cooldown Actuation on SIS Actuation;
- ESF 6.a: EFWS Actuation on Low-Low SG Level (All SGs);
- ESF 6.c. EEFWS Isolation on High SG Level (Affected SG); and
- ESF 8.a: MSIV Closure on SG Pressure Drop (All SGs).

SIGNAL PROCESSORS

1. Remote Acquisition Units

Two RAUs per division (8 total) are required to be OPERABLE when RTP is greater than or equal to 10%. These signal processors support the following functions:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop, and
- Reactor Trip 2: High Linear Power Density.

2. Acquisition and Processing Units

Five APUs per division (20 total) are required to be OPERABLE in accordance with the supported functions as shown in Table 3.3.1-2. These signal processors support the reactor trip, ESF functions, and Permissives.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Actuation Logic Units

Four ALUs per division (16 total) are required to be OPERABLE in MODES 1, 2, 3, 4, 5, 6, and during the movement of irradiated fuel assemblies. These signal processors support the reactor trip, ESF functions and Permissives.

ACTUATION DEVICES

1. RCP Bus and Trip Breakers

Two RCP Bus and Trip Breakers per pump (8 total) are required to be OPERABLE in MODES 1, 2, 3, and 4. These actuation devices support ESF 4: RCP Trip on Low Delta P across RCP with SIS.

2. Reactor Trip Circuit Breakers

Four RTCBs are required to be OPERABLE in MODES 1 and 2 and in MODE 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These actuation devices support the reactor trip functions.

3. Reactor Trip Contactors

Twenty three sets of four Reactor Trip Contactors (92 total) are required to be OPERABLE in MODES 1 and 2 and in MODE 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These actuation devices support the reactor trip functions.

ACTIONS

The most common causes of division inoperability are outright failure or drift of the sensor sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CALIBRATION when the sensor is set up for adjustment to bring it to within specification. ~~If the trip setpoint is non-conservative with respect to the Allowable Value, the division is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.~~ The SCP ensures that the divisions are performing as expected by confirming that the drift and other related errors are consistent with the supporting setpoint calculations.

BASES

ACTIONS (continued)

~~In the event a functions trip setpoint is found non-conservative with respect to the Allowable Value, or the sensors, signal processors, Actuation Signal Voting processors, or actuation devices are found inoperable, then all affected functions provided by that division must be declared inoperable, and the unit must enter any applicable Condition for the particular protection Function affected.~~

When the number of inoperable sensors or signal processors in a reactor trip or ESF function exceeds that specified in any related Condition, redundancy is lost and actions must be taken to restore the required redundancy.

A Note has been added to the ACTIONS. The Note has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each PS sensor, manual actuation switch, signal processor, and actuation device. The Completion Times of each inoperable sensor, manual actuation switch, signal processor, and actuation device will be tracked separately, starting from the time the Condition was entered for that sensor, manual actuation switch, signal processor, and actuation device.

A.1 and A.2

Condition A applies to the failure of one or more sensors. Condition A.1 applies only to the RTCB Position Indication sensors. If one or more of these sensors is inoperable, the inoperable sensor(s) must be placed in the tripped condition in 1 hour. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator action. Condition A.2 applies to all other PS sensors. If one or more of these sensors is inoperable, the inoperable sensor is placed in lockout in 4 hours. The 4 hour allotted timeframe is sufficient to allow the operator to take all appropriate actions for the failed sensor and still ensures that the risk involved in operating with the failure is acceptable.

B.1

Condition B applies to the failure of one or more manual actuation switches. In this condition, the minimum functional capability for manual actuation may not maintained. Restoring the manual initiation capability to OPERABLE status within 48 hours is reasonable considering the availability of automatic actuation, the low probability of an AOO or postulated accident occurring during this time, and the time necessary for repairs.

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies to one or more APUs inoperable due to the ~~LTSP~~ Setpoint Control Program requirements for one or more Trip/Actuation Functions not met. In this condition, the hardware is still functional. The sensors have been calibrated and the ADOTs and SOTs have checked the function from sensor to actuation device. The manual actuation capability would be unaffected. If the associated ~~inoperability affects the~~ Setpoint Control Program requirements are not met ~~LTSP~~ for either the EDG Start on Degraded Grid Voltage or the EDG Start on LOOP (Trip/Actuation Functions B.10.a or B.10.b), Required Action C.1 directs entry into the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown." The Completion Time of 1 hour is a reasonable time to allow the operator to diagnose and potentially correct the issue that caused the noncompliance with the associated Setpoint Control Program requirements ~~inoperability~~ prior to entering LCO 3.8.1 or LCO 3.8.2. Restoring compliance with the Setpoint Control Program requirements ~~LTSP to OPERABLE status~~ within 24 hours for all other Trip/Actuation Functions is a reasonable timeframe considering the time necessary to change the setpoint parameter, load corrected software, or replace the unit. If ~~the~~ compliance with the Setpoint Control Program requirements~~LTSP~~ cannot be restored ~~to OPERABLE status~~, the associated Trip/Actuation Function must be placed in lockout in the associated APU.

D.1 and D.2

Condition D applies to one or more signal processors inoperable for reasons other than Condition C. If the inoperability affects the APU associated with the EDG Start on Degraded Grid Voltage or the EDG Start on LOOP (Trip/Actuation Functions B.10.a or B.10.b), Required Action D.1 directs entry into the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown." The Completion Time of 1 hour is a reasonable time to allow the operator to diagnose and potentially correct the issue that caused the inoperability prior to entering LCO 3.8.1 or LCO 3.8.2. Restoring the Signal processor to OPERABLE status within 4 hours for all other Trip/Actuation Functions is a reasonable timeframe considering the time necessary to restore the signal processor to OPERABLE status. If the signal processor cannot be restored to OPERABLE status, the signal processor must be placed in lockout.

ACTIONS (continued)

E.1

Condition E applies to the RCP Bus and Trip Breakers, RTCBs, and Reactor Trip Contactors. With one ore more actuation devices inoperable, the actuation device must be restored to OPERABLE status within 48 hours. The Completion Time of 48 hours is reasonable considering that there are two automatic actuation divisions and the low probability of an event occurring during this interval.

F.1

If the Required Action and associated Completion Time of Condition A, B, C, D, or E or if the minimum functional capability (the value where the supported functions would not actuate during an AOO or postulated event coupled with a single failure) of the sensors, manual actuation switches, signal processors or actuation devices specified in Table 3.3.1-1 are not met, then the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE and any other specified actions must be taken.. The applicable Condition referenced in the table is sensor, manual actuation switch, signal processor, actuation device, and MODE dependent. Condition F is entered to provide for transfer to the appropriate subsequent Condition. Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1.

G.1

If Table 3.3.1-1 directs entry into Condition G, the unit must be brought to a condition in which the Low-Low RCS Loop Flow Rate in One Loop function (Trip/Actuation Function A.6.a) is not required to be OPERABLE. The allowed Completion Time of 2 hours is reasonable, based on operating experience, to reduce THERMAL POWER from full power to less than 70% in an orderly manner and without challenging unit systems.

H.1

If Table 3.3.1-1 directs entry into Condition H, the unit must be brought to a condition in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reduce THERMAL POWER from full power to less than 10% in an orderly manner and without challenging unit systems.

ACTIONS (continued)

I.1

If Table 3.3.1-1 directs entry into Condition I, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging unit systems.

J.1

If Table 3.3.1-1 directs entry into Condition J, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

K.1 and K.2

If Table 3.3.1-1 directs entry into Condition K, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and open the reactor trip breakers without challenging unit systems.

L.1 and L.2

If Table 3.3.1-1 directs entry into Condition L, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and then reduce the pressurizer pressure to less than 2005 psia without challenging unit systems.

BASES

ACTIONS (continued)

M.1 and M.2

If Table 3.3.1-1 directs entry into Condition M, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours to reach MODE 3 and 12 hours to reach MODE 4 is reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

N.1 and N.2

If Table 3.3.1-1 directs entry into Condition N, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours to reach MODE 3 and 36 hours to reach MODE 5 is reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

O.1

If Table 3.3.1-1 directs entry into Condition O, the Conditions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, "AC Sources - Shutdown," for the EDG made inoperable by failure of the 6.9 kV Bus Voltage sensors are required to be entered immediately. The actions of those LCOs provide adequate compensatory actions to assure unit safety.

P.1

If Table 3.3.1-1 directs entry into Condition P, the associated CVCS isolation valves are immediately declared inoperable. The actions of LCO 3.4.9, "Pressurizer," provide adequate compensatory actions to assure unit safety.

Q.1

If Table 3.3.1-1 directs entry into Condition Q, the associated PSRVs are immediately declared inoperable. The actions of LCO 3.4.10, "Pressurizer Safety Relief Valves," provide adequate compensatory actions to assure unit safety.

BASES

ACTIONS (continued)

R.1

If Table 3.3.1-1 directs entry into Condition R, both Control Room Emergency Filtration trains are immediately declared inoperable. The actions of LCO 3.7.10, "Control Room Emergency Filtration (CREF)," provide adequate compensatory actions to assure unit safety.

S.1

If Table 3.3.1-1 directs entry into Condition S, the manual Reactor Trip switches are inoperable. If the switches cannot be returned to OPERABLE status within one hour, actions must be taken to ensure all RCCAs are inserted and the reactor must be placed in a condition where the RCCA can not be withdrawn. This is accomplished by opening the reactor trip breakers. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator action.

T.1

If Table 3.3.1-1 directs entry into Condition T, the associated ALUs must be immediately declared inoperable. If the ALUs cannot be returned to OPERABLE status within one hour, actions must be taken to ensure all RCCAs are inserted and the reactor must be placed in a condition where the RCCA can not be withdrawn. This is accomplished by opening the reactor trip breakers. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator action.

SURVEILLANCE REQUIREMENTS

The SRs for any particular PS sensor, manual actuation switch, signal processor, or actuation device are found in the SR column of Table 3.3.1-1 for that sensor, manual actuation switch, signal processor, or actuation device.

~~REVIEWER'S NOTE~~

~~In order for a plant to take credit for topical reports as the basis for justifying Frequencies, topical reports must be supported by an NRC staff SER that establishes the acceptability of each topical report for that unit.~~

~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~------------------------------- REVIEWER'S NOTE --------------------------------~~
~~The Notes in Table 3.3.1-1 requiring reset of the division to a predefined as-left tolerance and the verification of the as-found tolerance are only associated with SL-LSSS values. Therefore, the Notes are placed at the top of the LTSP column in the Table and applied to all Functions with LTSPs in the table. The Notes may be applied to specific SRs for the associated functions in the SR column only.~~
~~---~~

~~------------------------------- REVIEWER'S NOTE --------------------------------~~
~~Notes b and c are applied to the setpoint verification Surveillances for all SL-LSSS Functions unless one or more of the following exclusions apply:~~

1. ~~Notes b and c are not applied to SL-LSSS Functions which utilize mechanical components to sense the trip setpoint or to manual initiation circuits (the latter are not explicitly modeled in the accident analysis). Examples of mechanical components are limit switches, float switches, proximity detectors, manual actuation switches, and other such devices that are normally only checked on a "go/no go" basis. Note 1 requires a comparison of the periodic surveillance requirement results to provide an indication of Trip/Actuation Function (or individual device) performance. This comparison is not valid for most mechanical components. While it is possible to verify that a limit switch functions at a point of travel, a change in the surveillance result probably indicates that the switch has moved, not that the input/output relationship has changed. Therefore, a comparison of surveillance requirement results would not provide an indication of the Trip/Actuation Function or component performance.~~

2. ~~Notes b and c are not applied to Technical Specifications associated with mechanically operated safety relief valves. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Section XI testing program.~~

3. ~~Notes b and c are not applied to SL-LSSS Functions and Surveillances which test only digital components. For purely digital components, such as actuation logic circuits and associated relays, there is no expected change in result between surveillance performances other than measurement and test errors (M&TE) and, therefore, comparison of Surveillance results does not provide an indication of Trip/Actuation Function or component performance.~~

~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~An evaluation of the potential SL-LSSS Functions resulted in Notes b and c being applied to the Functions shown in the TS markups. Each licensee proposing to fully adopt this TSTF must review the potential SL-LSSS Functions to identify which of the identified functions are SL-LSSS according to the definition of SL-LSSS and their plant specific safety analysis. The two TSTF Notes are not required to be applied to any of the listed Functions which meet any of the exclusion criteria or are not SL-LSSS based on the plant specific design and analysis.~~

~~The Limiting Trip Setpoint column for reactor trip functions is modified by two Footnotes as identified in Table 3.3.1-2. The selected Functions are those Functions that are LSSS for protection system instrument functions that protect reactor core or RCS pressure boundary SLs. Some components (e.g., mechanical devices which have an on or off output or an open/close position such as limit switches, float switches, and proximity detectors) are not calibrated in the traditional sense and do not have as-left or as-found conditions that would indicate drift of the component setpoint. These devices are considered not trendable and the requirements of the Notes are not required to be applied to the mechanical portion of the functions. Where a non-trendable component provides signal input to other Trip/Actuation Function components that can be trended, the remaining components must be evaluated in accordance with the Notes.~~

~~The first Note requires evaluation of Trip/Actuation Function's performance for the condition where the as-found setting for the setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. For digital channel components, the as-found tolerance may be identical to the as-left tolerance since drift may not be an expected error. In these cases a Trip/Actuation Function's as-found value outside the as-left condition may be cause for component assessment. Evaluation of instrument performance will verify that the instrument will continue to behave in accordance with design-basis assumptions. The purpose of the assessment is to ensure confidence in the instrument performance prior to returning the instrument to service. These conditions will also be identified in the Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for continued OPERABILITY.~~

~~The second Footnote requires that the as-left setting for the instrument be returned to within the as-left tolerance of the LTSP. Where a setpoint more conservative than the LTSP is used in the plant surveillance procedures, the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~sufficient margin to the SL and/or Analytical Limit is maintained. If the as-left instrument setting cannot be returned to a setting within the as-left tolerance, then the Trip/Actuation Function shall be declared inoperable. The second Note also requires that the LTSP and the methodologies for calculating the as-left and the as-found tolerances be in a document controlled under 10 CFR 50.59.~~

The digital PS provides continual online automatic monitoring of each of the input signal in each division, perform software limit checking (signal online validation) against required acceptance criteria, and provide hardware functional validation so that a division check is continuously being performed. If any PS input signal is identified to be in a failure status, this condition is alarmed in the Control Room. As such, a periodic "channel check" is no longer necessary.

SR 3.3.1.1

SR 3.3.1.1~~2~~ compares the calorimetric heat balance calculation to the power range division output every 24 hours. If the calorimetric heat balance calculation results exceed the power range division output by more than 2% RTP, the power range division is not declared inoperable, but must be adjusted. The power range division output shall be adjusted consistent with the calorimetric heat balance calculation results if the calorimetric calculation exceed the power range division output by more than + 2% RTP. If the power range division output cannot be properly adjusted, the division I is declared inoperable.

If the calorimetric is performed at part power (< 70% RTP), adjusting the power range division indication in the increasing power direction will assure a reactor trip below the safety analysis limit (< 11% RTP). Making no adjustment to the power range division in the decreasing power direction due to a part power calorimetric assures a reactor trip consistent with the safety analyses.

This allowance does not preclude making indicated power adjustments, if desired, when the calorimetric heat balance calculation is less than the power range division output. To provide close agreement between indicated power and to preserve operating margin, the power range divisions are normally adjusted when operating at or near full power during steady-state conditions. However, discretion must be exercised if the power range division output is adjusted in the decreasing power direction due to a part power calorimetric (< 70% RTP). This action may introduce a non-conservative bias at higher power levels. The cause of

SURVEILLANCE REQUIREMENTS (continued)

the potential non-conservative bias is the decreased accuracy of the calorimetric at reduced power conditions. The primary error contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement, which is typically a delta P measurement across a feedwater venturi. While the measurement uncertainty remains constant in delta P as power decreases, when translated into flow, the uncertainty increases as a square term. Thus a 1% flow error at 100% power can approach a 10% flow error at 30% RTP even though the delta P error has not changed. An evaluation of extended operation at part power conditions would conclude that it is prudent to administratively adjust the setpoint of the High Neutron Flux Rate of Change when: 1) the power range division output is adjusted in the decreasing power direction due to a part power calorimetric below 70% RTP; or 2) for a post refueling startup. The evaluation of extended operation at part power conditions would also conclude that the potential need to adjust the indication of the High Neutron Flux Rate of Change in the decreasing power direction is quite small, primarily to address operation in the intermediate range about 10% RTP to allow enabling of the High Neutron Flux Rate of Change reactor trips. Before the High Neutron Flux Rate of Change setpoint is reset, the power range division adjustment must be confirmed based on a calorimetric performed at ≥ 70% RTP.

The Note clarifies that 12 hours are allowed for performing the first Surveillance after reaching 20% RTP. A power level of 20% RTP is chosen based on plant stability, (i.e., automatic rod control capability and turbine generator synchronized to the grid). The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range division output of more than +2% RTP is not expected in any 24 hour period.

SR 3.3.1.2

Space- and time- dependent power density distribution of the U.S. EPR is accurately assessed using the SPNDs inside the core. For neutron flux measurement, incore neutron detectors are more accurate than excore neutron detectors. CALIBRATION of SPND instrumentation is performed to compensate for a decrease in SPND sensitivity during the fuel cycle and to account for peak power density factor change over the fuel cycle. The Aeroball Measurement System (AMS) assists in generating the measured relative neutron flux density in the core, which is used in

BASES

SURVEILLANCE REQUIREMENTS (continued)

conjunction with the predicted power distribution based on actual core operation to calibrate the incore SPND instrumentation. Because both the power-to-signal ratio of an SPND and the reference power distribution change with core burnup, SPND signals are matched to reference signals provided by the AMS every 15 EFPD (Ref. 7).

The Note clarifies that 12 hours are allowed for performing the first Surveillance after reaching 20% RTP. A power level of 20% RTP is chosen based on plant stability, (i.e., automatic rod control capability and turbine generator synchronized to the grid).

SR 3.3.1.3

SR 3.3.1.3 is the performance of a ADOT every 31 days. This test shall verify OPERABILITY by actuation of the Reactor Trip Circuit Breakers and Reactor Trip Contactors. The ADOT may be performed by means of any series of sequential, overlapping, or total steps.

SR 3.3.1.4

The online boron meters are a half shell design and are not in contact with the reactor coolant. The concentration of boron is measured by using the neutron absorption effect of B^{10}. The boron concentration is calculated using the measured count rate. To improve the accuracy of the measurement, the temperature of the reactor coolant at the measuring point is used to adjust the boron concentration. The temperature instruments are not included as part of this Surveillance. The frequency of the boron meter CALIBRATION is conservative considering instrument reliability.

Specification 5.5.18.a requires that the Limiting Trip Setpoint (LTSP), Allowable Value (AV), as-found tolerance (AFT), and the as-left tolerance (ALT), as well as the methodology for calculating these be in the Setpoint Control Program (SCP).

The SCP provides requirements for the calibration reset and evaluation of the performance of required divisions. As indicated in Specification 5.5.18.c.1 evaluation of division performance is required for the condition where the "as-found" setting for the division is outside its AFT, but conservative with respect to the AV. Evaluation of division performance will verify that the instrument will continue to behave in accordance with design-basis assumptions. The purpose of the assessment is to ensure confidence in the instrument performance prior to returning the instrument to service. These divisions will also be identified in the Corrective Action

BASES

SURVEILLANCE REQUIREMENTS (continued)

Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for OPERABILITY. For digital division components and Functions whose instruments are mechanical devices (e.g., devices which have an "on" or "off" output or an open/close position such as limit switches, float switches, and proximity detectors), the AFT may be identical to the ALT because drift may not be an expected error.

As indicated in Specification 5.5.18.c.2, the as-left setting for the instrument is required to be returned to within the ALT around the LTSP. Where a setpoint more conservative than the LTSP is used in plant surveillance procedures, the ALT and AFT, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the AV is maintained. If the as-left instrument setting cannot be returned to a setting within the ALT, then the instrument division shall be declared inoperable.

SR 3.3.1.5

A SOT on each required reactor trip actuation device is performed every 24 months to ensure the devices will perform their intended function when needed. A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for division OPERABILITY. The SOT shall include the verification of the accuracy and time constants of the analog input modules. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint methodology as required by the SCP.

The maximum permissible response time for analog input modules is prescribed by the process engineering of the specific application. Thus for each applicable PS function, the limiting response times will be shown to be consistent with the safety requirements.

The response time testing is performed in overlapping steps:

- Verification of time constants of the input divisions during input module tests, and
- Verification of the signal propagation time within the digital system.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The response time of the analog input divisions are tested periodically by injection of test signals in the input circuits. For this purpose, an external test computer is temporarily connected to the I&C system via permanently installed test plugs. While the input from the process is deactivated (by switching off the associated division(s) power supply), a binary input is provided to the data acquisition computers. The signal distribution to other computers is designed in the application software in the same way as for the normal measuring signals. Separate outputs are provided in the voting computers for each path. During the response time tests, the test machine connected to the I&C system generates a start signal and measures the reaction time of each signal path separately to verify that it does not exceed the worst case conditions specified for the specific system configuration. The measurements are performed a number of times to determine the statistical characteristics of each signal path.

The SOT may be performed by means of any series of sequential, overlapping, or total steps.

SR 3.3.1.6

A CALIBRATION of each PS sensor (except neutron detectors) every 24 months ensures that each instrument division is reading accurately and within tolerance. A CALIBRATION shall be the adjustment, as necessary, of the sensor output such that it responds within the necessary range and accuracy to known values of the parameter that the sensor monitors. The CALIBRATION shall encompass all devices in the division required for sensor OPERABILITY. CALIBRATION of instrument divisions with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal CALIBRATION of the remaining adjustable devices in the division. The CALIBRATION may be performed by means of any series of sequential, overlapping, or total steps.

Specification 5.5.18.a requires that the Limiting Trip Setpoint (LTSP), Allowable Value (AV), as-found tolerance (AFT), and the as-left tolerance (ALT), as well as the methodology for calculating these be in the Setpoint Control Program (SCP).

The SCP provides requirements for the calibration reset and evaluation of the performance of required divisions. As indicated in Specification 5.5.18.c.1 evaluation of division performance is required for the condition where the "as-found" setting for the division is outside its AFT, but conservative with respect to the AV. Evaluation of division performance will verify that the instrument will continue to behave in accordance with design-basis assumptions. The purpose of the assessment is to ensure

BASES

SURVEILLANCE REQUIREMENTS (continued)

confidence in the instrument performance prior to returning the instrument to service. These divisions will also be identified in the Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for OPERABILITY. For digital division components and Functions whose instruments are mechanical devices (e.g., devices which have an "on" or "off" output or an open/close position such as limit switches, float switches, and proximity detectors), the AFT may be identical to the ALT because drift may not be an expected error.

As indicated in Specification 5.5.18.c.2, the as-left setting for the instrument is required to be returned to within the ALT around the LTSP. Where a setpoint more conservative than the LTSP is used in plant surveillance procedures, the ALT and AFT, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the AV is maintained. If the as-left instrument setting cannot be returned to a setting within the ALT, then the instrument division shall be declared inoperable.

SR 3.3.1.7

The features of continuous self-monitoring of the PS system are described in Reference 8. Additional tests, which require the processor to be inoperable are not normally performed during operation. These EXTENDED SELF TESTS are performed at start-up of a computer each cycle. The startup sequence is as follows:

- Hardware basic test using the internal diagnosis monitor,
- Start-up self test of the operating system, and
- Switch over to normal operation after approximately two minutes.

Additional information is provided in Section 3 of Reference 8.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a ADOT every 31 days. This test shall verify OPERABILITY by actuation of the RCP Bus and Trip Breakers. The ADOT may be performed by means of any series of sequential, overlapping, or total steps.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.9

SR 3.3.1.9 verifies that the ~~Limiting Trip S~~setpoint and p~~P~~ermissive values have been properly loaded into the applicable APU.

REFERENCES

1. ANP-10275P, Revision 0, U.S. EPR Instrument Setpoint Methodology Topical Report, March 2007.
2. 10 CFR 100.
3. 10 CFR 50, Appendix A, GDC 21.
4. ANP-10287, Incore Trip Setpoint and Transient Methodology for U.S. EPR, November 2007.
5. FSAR Chapter 15.
6. 10 CFR 50.49.
7. ANP-10271P, Revision 0, US EPR Nuclear Incore Instrumentation Systems Report, December 2006.
8. EMF-2341(P), Revision 1, Generic Strategy for Periodic Surveillance Testing of TELEPERM XS System in U.S. Nuclear Generating Stations, March 2000.

Table B 3.3.1-1 (page 1 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
A. REACTOR TRIPS						
1. Low Departure from Nucleate Boiling Ratio (DNBR) a. Low DNBR d. High Quality	≥ 10% RTP	3	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit

Table B 3.3.1-1 (page 2 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
1. Low Departure from Nucleate Boiling Ratio (DNBR) b. Low DNBR and (Imbalance or Rod Drop) c. Variable Low DNBR and Rod Drop e. High Quality and Imbalance or Rod Drop	≥ 10% RTP	3	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit

Table B 3.3.1-1 (page 3 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
2. High Linear Power Density	≥ 10% RTP	3	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit
4. High Core Power Level	1,2[(a)]	3	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit

(a) ≥ 10-5% power on the intermediate range detectors.

Table B 3.3.1-1 (page 4 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
5. Low Saturation Margin	1,2[(a)]	3	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit

(a) ≥ 10-5% power on the intermediate range detectors.

Table B 3.3.1-1 (page 5 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY	SENSORS / PROCESSORS DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
B. ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) SIGNALS						
2.e. Main Feedwater / Startup and Shutdown Feedwater Isolation on Steam Generator Level High for Period of Time (Affected Steam Generators)	1,2[b],3[b]	3	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit
3.b. ESF - Safety Injection System (SIS) Actuation on Low Delta P_{sat}	3[c]	3	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit

(b) Except when all MFW low load isolation valves are closed.

(c) When Trip/Actuation Function B.3.a, SIS Actuation on Low Pressurizer Pressure, is disabled.

Table B 3.3.1-1 (page 6 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
4. ESF - Reactor Coolant Pump (RCP) Trip on Low Delta P across RCP with Safety Injection System Actuation	1,2,3	3	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit
11a. ESF - Chemical and Volume Control System (CVCS) Charging Line Isolation on High-High Pressurizer Level	1,2,3	3	Pressurizer Level / Acquisition and Processing Unit	Pressurizer Level / Acquisition and Processing Unit	Pressurizer Level / Acquisition and Processing Unit	Pressurizer Level / Acquisition and Processing Unit
	1,2,3	2	Actuation Logic Unit (1 of 2)			Actuation Logic Unit (1 of 2)

Table B 3.3.1-1 (page 7 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
11b. ESF - Chemical and Volume Control System (CVCS) Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating)	5[(d)],6	3	Boron Concentration Boron Temperature / Acquisition and Processing Unit	Boron Concentration Boron Temperature / Acquisition and Processing Unit	Boron Concentration Boron Temperature / Acquisition and Processing Unit	Boron Concentration Boron Temperature / Acquisition and Processing Unit
	5[(d)],6	2	Actuation Logic Unit (1 of 2)			Actuation Logic Unit (1 of 2)
11c. ESF - Chemical and Volume Control System (CVCS) Charging Line Isolation on ADM at Standard Shutdown Conditions	3,4[(e)],5[(e)]	3	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit
	3,4[(e)],5[(e)]	2	Actuation Logic Unit (1 of 2)			Actuation Logic Unit (1 of 2)

(d) With two or less RCPs in operation.

(e) With three or more RCPs in operation.

B 3.3 INSTRUMENTATION

B 3.3.2 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to provide operators with information that is needed during accidents.

The OPERABILITY of PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following accidents and transients when the use of the Emergency Operating Procedures (EOPs) is required.

The PAM instruments included in Table 3.3.2-1, Postaccident Monitoring Instrumentation, are required for the following reasons:

1. Perform the diagnosis specified in the emergency operating procedures (these variables are restricted to preplanned actions for the primary success path of DBAs), e.g., loss of coolant accident (LOCA);

2. Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;

3. Provide information to indicate whether plant safety functions are being accomplished for reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity (including radioactive effluent control);

4. Provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases (i.e., fuel cladding, primary coolant pressure boundary, and containment); and

5. Enable the operator to recognize which heat transfer symptom is occurring: 1) loss of subcooling margin, 2) lack of heat transfer, 3) excessive heat transfer, and 4) Steam Generator Tube Rupture.

The PAM instrumentation is displayed through the Safety Information and Control Systems (SICS), which includes the Qualified Display System (QDS). The Safety Automation System (SAS) communication with the QDS (as part of SICS) is realized through the Monitoring and Service Interfaces (MSI), and the Panel Interfaces (PI). The PI's are part of the SICS, the MSI's are part of the SAS. The SAS also provides outputs for analog meters, illuminated buttons etc., and receives inputs from Conventional Instrumentation and Controls which is included in the SICS.

BASES

BACKGROUND (continued)

The SICS calculates a margin to saturation by using the safety grade inputs of RCS Hot Leg Pressure, RCS Hot Leg Temperature, and Incore Thermocouples. The margin, both positive and negative, is available for diagnosis of plant transients. As long as adequate subcooling margin exists, core cooling is ensured. If subcooling margin is lost, actions are required to ensure core cooling and restore adequate subcooling margin. Superheat is used for Inadequate Core Cooling (ICC) determination and initiation of more severe mitigation guidance to restore saturated and ultimately subcooled coolant conditions.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the OPERABILITY of certain Regulatory Guide 1.97 variables, so that the control room operating staff can:

- Recognize when a heat transfer symptom is occurring that would require performance of the appropriate section in the emergency operating procedures.
- Perform the diagnosis specified in the emergency operating procedures (these variables are restricted to preplanned actions for the primary success path of postulated accidents), e.g., loss of coolant accident (LOCA);
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;
- Determine whether systems important to safety are performing their intended functions for reactivity control, core cooling, maintaining reactor coolant system integrity and maintaining containment integrity,
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

BASES

APPLICABLE SAFETY ANALYSES (continued)

PAM instrumentation used to support pre-planned, manually controlled actions satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii)(C). The other PAM instrumentation that perform certain functions related to verification of key safety functions and monitoring key barriers for potential breach must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, these variables are important for reducing public risk.

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 monitors that provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that perform certain functions related to verification of key safety functions and monitoring key barriers for potential breach.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident.

LCO 3.3.2 requires two OPERABLE divisions for most Functions. Two OPERABLE divisions ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident.

Furthermore, OPERABILITY of two divisions allows for a comparison during the post accident phase to confirm the validity of displayed information.

The exception to the two division requirement is Penetration Flow Path Containment Isolation Valve (CIV) Position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

BASES

LCO (continued)

PAM variables are required to meet design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.2-1.

1. Cold Leg Temperature (Wide Range)

The key variables for monitoring core cooling are Hot Leg Temperature, Core Exit Temperature, and Steam Generator Pressure. Cold Leg Temperature provides backup temperature monitoring to Hot Leg Temperature and Core Exit Temperature when forced or verified natural circulation exists. Cold Leg Temperature is used with Hot Leg Temperature and Core Exit Temperature to verify natural circulation. Cold Leg Temperature is compared to the saturation temperature for steam generator pressure (Tsat) to determine primary to secondary loop coupling.

2. Containment Isolation Valve Position Indication

Containment isolation valve position verifies Containment isolation and is required to ensure Containment integrity in event of a LOCA.

3. Containment Pressure

Containment pressure is a key measurement used for detection of a LOCA, verification of Engineered Safety Features mitigation, and detection of a potential breach of Containment.

4. Emergency Feedwater Storage Pool Level

Emergency feedwater pool level is a key variable to ensure adequate EFW pump net positive suction head (NPSH) is satisfied.

BASES

LCO (continued)

5. Emergency Feedwater System Flow

Emergency Feedwater flow indication is required when throttling feedwater flow to the steam generators. Control of flow is required to control the rate of steam generator heat removal to maintain Reactor Coolant temperature profiles within limits for cooldown.

6. Extra Boration System Flow

The Extra Boration System flow provides verification that the appropriate system alignment has been completed. The negative reactivity additions performed by this system require verfication of correct system operation.

7. Hot Leg Injection Flow

Hot Leg Injection flow provides verification that the appropriate system alignment has been completed. Hot leg injection is required to prevent the buildup of sufficient boron concentration in the core coolant channels to impede long term core cooling.

8. Hot Leg Pressure (Wide Range)

RCS pressure is required to monitor reactor coolant integrity and assess core cooling. RCS pressure and either RCS hot leg or incore temperature is used to determine subcooling margin if the calcuation is not available.

9. Hot Leg Temperature (Wide Range)

Hot Leg Temperature is required to monitor core cooling, to verify natural circulation, and to verify primary to secondary loop coupling along with steam generator pressure. Hot Leg temperature and RCS pressure are used to determine loop subcooling margin if the calculation is not available.

BASES

LCO (continued)

10. In-containment Refueling Water Storage Tank Level

In-Containment storage tank level is monitored during operation to ensure that adequate pump NPSH is maintained during the recirculation phase of LOCA mitigation for long term core cooling requirements. In addition, level instrumentation is used to assess level loss due to leakage on Safety Injection piping located outside of Containment and interfacing systems (Inter-system LOCA) as well as level rise due to dilution mechanisms.

11. Incore Temperature

Core cooling is monitored by RCS and incore thermocouple temperatures. Loss of subcooling margin (SCM) is identified by a combination of RCS pressure and either hot leg temperature or incore thermocouple temperature depending on plant conditions, (e.g., RCPs on/off). Incore Temperature is monitored for verification and surveillance of long term core cooling and to detect potential breach of fuel cladding.

12. Power Range Monitors

Power Range Neutron Flux is used to verify that reactor trip has resulted in "Reactor Shutdown". Once "Reactor Shutdown" is verified following reactor trip, all subsequent EOP action is based on a shutdown reactor. Power Range indication is used during a steam generator tube rupture to determine when core power is within the Main Steam bypass capability, at which point a reactor trip can be performed without challenge to the Main Steam relief capabilities.

13. Pressurizer Level

Pressurizer level provides information for the operator to maintain RCS pressure and inventory control, with the exception of a few accident situations, such as large break LOCA. Pressurizer level is a key variable required to ensure proper operation of the pressurizer heaters to maintain the pressurizer in a saturated state.

BASES

LCO (continued)

14. Radiation Monitor - Containment High Range Activity

Containment high range radiation instrumentation is used to assess the potential for significant radiation releases and to provide release assessment for determining the need to invoke site emergency plans.

15. Radiation Monitor - Main Steam Line Activity

Main Steam Line radiation levels are a key variable for detection of a breach between the primary to secondary loop boundary.

16. Source Range Monitors

Source Range instrumentation is used to ensure that the reactor remains subcritical. Once "Reactor Shutdown" is verified following reactor trip, all subsequent EOP action is based on a shutdown reactor. Source range can be used to assess if a return to critical condition is approached during plant cooldown and whether mitigation efforts are required to maintain the reactor in a shutdown condition.

17. Steam Generator Level (Wide Range)

Both steam generator level and pressure are monitored to assess primary to secondary heat transfer. An upper level limit is specified to prevent moisture carryover into the steam lines which could damage control components used for controlling RCS cooldown.

18. Steam Generator Pressure

Steam Generator pressure is a key parameter used to identify upsets in heat transfer and evaluate primary-to-secondary heat transfer.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate postulated accidents. The applicable postulated accidents are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, plant conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

BASES

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.2-1. The Completion Time(s) of the inoperable division(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

When one or more Functions have one required division that is inoperable, the required inoperable division must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE division (or in the case of a Function that has only one required division, other non-Regulatory Guide 1.97 instrument divisions to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

This Required Action specifies initiation of actions in accordance with Specification 5.6.5, which requires a written report to be submitted to the Nuclear Regulatory Commission. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative Required Actions. This Required Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Required Actions are identified before a loss of functional capability condition occurs.

C.1

When one or more Functions have two required divisions inoperable (i.e., two divisions inoperable in the same Function), one division in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of alternate means to obtain the required information. Continuous operation with two required divisions inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable division of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

BASES

ACTIONS (continued)

D.1 and D.2

If the Required Action and associated Completion Time of Condition C are not met and Table 3.3.2-1 directs entry into Condition E, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

A Note at the beginning of the SR Table specifies that the following SR applies to each PAM instrumentation Function found in Table 3.3.2-1.

SR 3.3.2.1

A CALIBRATION is performed every 24 months or approximately every refueling. CALIBRATION is a complete check of the instrument division including the sensor. The Surveillance verifies the function responds to the measured parameter within the necessary range and accuracy. A Note allows exclusion of the neutron detectors from the CALIBRATION. The requirements for CALIBRATION of neutron detectors is Specified in Specification 3.3.1," Protection System and Safety Automation System".

The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of a 24 month CALIBRATION interval for the determination of the magnitude of equipment drift.

SR 3.3.2.2

A SOT on each Safety Information and Control System performing the PAM functions listed in Table 3.3.2-1 is performed every 24 months to ensure the entire division will perform its intended function when needed. A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for division OPERABILITY. The SOT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for division OPERABILITY such that the setpoints are within the necessary range and accuracy. The SOT may be performed by means of any series of sequential, overlapping, or total steps.

BASES

REFERENCES

1. NUREG 0737, Supplement 1.

B 3.3 INSTRUMENTATION

B 3.3.3 Remote Shutdown System (RSS)

BASES

BACKGROUND

The RSS provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as Hot Standby (MODE 3). With the unit in MODE 3, the Emergency Feedwater (EFW) System and Main Steam Relief Train (MSRT) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the EFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allow extended operation in MODE 3.

The RSS contains Human Machine Interface (HMI) workstations necessary to bring the plant to and maintain it in a safe shutdown state. The HMI (control) functions of the RSS are isolated as long as the Main Control Room (MCR) is available. The HMI workstations will continue to display all parameters available on each workstation while the control functions are isolated. These workstations contain Process Information and Control System (PICS) equipment, Safety Information and Control System (SICS) equipment, and select communication equipment.

In the event that the control room becomes inaccessible, the operators can establish control at the remote shutdown panel and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are required to be located at the remote shutdown panel. Some controls and transfer switches may be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the RSS control and instrumentation Functions ensures that there is sufficient information available on selected plant parameters to bring the plant to, and maintain it in, MODE 3 should the control room become inaccessible.

BASES

APPLICABLE SAFETY ANALYSES

The RSS is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down the plant and maintain it in a safe condition in MODE 3.

The criteria governing the design and the specific system requirements of the RSS are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

The RSS satisfies Criterion 4 of 10 CFR 50.36(d)(2)(ii).

LCO

The RSS LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the plant in MODE 3 from a location other than the control room. The instrumentation and controls required are listed in Table B 3.3.3-1.

The controls, instrumentation, and transfer switches necessary to reach MODE 3 are those required for:

- Reactivity Control (initial and long term),
- Reactor Coolant Make-up
- RCS Pressure Control,
- Decay Heat Removal, and
- Safety support systems for the above Functions, as well as service water, component cooling water, and onsite power including the Emergency Diesel Generators.

The systems are controlled by the Safety Automation System (LCO 3.3.1, “Protection System and Safety Automation System”).

A Function of a RSS is OPERABLE if all instruments and controls needed to support the Remote Shutdown System Function are OPERABLE.

The remote shutdown instrument and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments and control circuits will be OPERABLE if unit conditions require that the RSS be placed in operation.

BASES

APPLICABILITY

The RSS LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the unit is already subcritical and in the condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument control Functions if control room instruments or control become unavailable.

ACTIONS

A RSS division is inoperable when each Function is not accomplished by at least one designated RSS division that satisfies the OPERABILITY criteria for the division's Function. These criteria are outlined in the LCO section of the Bases.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Time(s) of the inoperable division(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

<u>A.1</u>

Condition A addresses the situation where one or more functions of the RSS are inoperable. This includes the control and transfer switches for any required Function.

The Required Action is to restore the divisions to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

<u>B.1 and B.2</u>

If the Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

SR 3.3.3.1 verifies that each required RSS transfer switch and control circuit performs its intended function. This verification is performed from the reactor shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. Displays in the MCR and RSS contain real-time plant data prior to, during, and after control transfer from one station to the other. The RSS data is populated from the same information busses that supply data to the MCR. During the time control is transferred from the MCR to the RSS or vice versa, the operator will have seamless transfer of control and data will not be interrupted. The operators will have an indication via the control system that RSS control has been established. This will ensure that if the control room becomes inaccessible, the plant can be brought to and maintained in MODE 3 from the reactor shutdown panel and the local control stations. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience demonstrates that RSS control usually pass the Surveillance when performed at a Frequency of once every 24 months.

SR 3.3.3.2

A CALIBRATION of each instrument display function on the RSS every 24 months ensures that each instrument division is reading accurately and within tolerance. A CALIBRATION is a complete check of the instrument division, including the sensor. The test verifies that the division responds to the measured parameter within the necessary range and accuracy. CALIBRATION leaves the division adjusted to account for instrument drift to ensure that the division remains operational between successive tests.

SR 3.3.3.3

A SOT on each division performing the RSS functions is performed every 24 months to ensure the entire division will perform its intended function when needed. A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for division OPERABILITY. The SOT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for division OPERABILITY such that the setpoints are within the necessary range and accuracy. The SOT may be performed by means of any series of sequential, overlapping, or total steps.

BASES

REFERENCES	1.	10 CFR 50, Appendix A, GDC 19.

Table B 3.3.3-1 (page 1 of 2)
Remote Shutdown System Instrumentation and Controls

FUNCTION / INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF FUNCTIONS
Source Range Neutron Flux	1
Control Rod Drive Mechanism (CRDM) Bottom Position Indications	1 per CRDM
Reactor Trip Breakers	1 per trip breaker
Reactor Coolant Pump Trip	1 per pump
RCS Hot Leg Pressure Wide Range	1 per loop
RCS Hot Leg Temperature (Wide Range)	1 per loop
RCS Cold Leg Temperature (Wide Range)	1 per loop
Pressurizer Pressure	1
Pressurizer Pressure Setpoint Reset	1
Low Temperature Overpressure Alarm	1
Pressurizer Level	1
Pressurizer Level Variable Setpoints	1
Pressurizer Safety Relief Valves (incl. Actuators and Position Sensors	1 per valve
Steam Generator Pressure	1 per loop
Steam Generator Pressure Variable Setpoints	1 per loop
Steam Generator Pressure Setpoint Reset	1 per loop
Steam Generator (Wide Range) Levels	1 per loop
Main Steam Isolation Valves	1 per valve
Main Steam Relief Isolation Valves	1 per valve
Main Steam Relief Control Valves	1 per valve
In Containment Refueling Water Storage Tank (IRWST) Level	1
Low Head Safety Injection (LHSI) Pumps	1 per pump
Residual Heat Removal (RHR) Heat Exchanger Main Control Valves	1 per loop
RHR Heat Exchanger Bypass Control Valves	1 per loop
RHR Heat Exchanger Inlet Temperatures	1 per loop
RHR Heat Exchanger Outlet Temperatures	1 per loop
RHR Suction Line Isolation Valves	1 per loop
RHR Suction Line Isolation Valve Interlock Status	1 per loop
RHR Warm-Up / Conditioning Valves	1 per loop
Essential Service Water Pumps	1 per loop
Component Cooling Water (CCW) Pumps	1 per pump
CCW Surge Tank Level	1 per tank
Emergency Diesel Generator (EDG)	1 per EDG
CVCS Letdown Isolation Valves	1 per valve
EBS Boric Acid Storage Tank Levels	1
EBS Injection Line Isolation Valves	1 per valve
EBS Pumps	1 per pump

Table B 3.3.3-1 (page 2 of 2)
Remote Shutdown System Instrumentation and Controls

FUNCTION / INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF FUNCTIONS
EBS Containment Isolation Valves	1 per valve
Emergency Feedwater Pumps	1 per pump
Emergency Feedwater Pool Levels (WR)	1 per pool
P12 Permissive	1
P14 Permissive	1
P15 Permissive	1
P17 Permissive	1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of a cylindrical reinforced concrete outer Shield Building, a cylindrical post-tensioned concrete inner Containment Building with a 0.25-inch thick steel liner, and an annular space between the two buildings. The containment, including all its penetrations, is a low leakage shell designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the containment provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

An annular space exists between the walls and domes of the Containment Building and the Shield Building to permit inservice inspection and collection of containment outleakage.

Containment piping penetration assemblies provide for the passage of process, service, sampling and instrumentation pipelines into the reactor vessel while maintaining containment OPERABILITY. The Shield Building allows controlled release of the annulus atmosphere under accident conditions, as well as protecting the Containment Building from external hazards.

The inner Containment Building and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

    1. Capable of being closed by an OPERABLE automatic containment isolation system; or

    2. Closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves."

BASES

BACKGROUND (continued)

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks;" and

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA, a main steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting Design Basis Accident. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25% per day in the safety analysis at P_a = 55 psig (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

BASES

LCO (continued)

Individual leakage rates specified for the containment air locks (LCO 3.6.2) and purge valves with resilient seals are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 La.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as containment post tensioning surveillance, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required Containment Leakage Rate Testing Program leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $1.0\ L_a$. At $1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

~~-----------------------------------REVIEWER' NOTE-----------------------------------~~
~~Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.~~
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SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Post Tensioning Surveillance Program. Testing and Frequency are in accordance with the ASME Code, Section III, Division 2, 2004 (Ref. 4).

BASES

REFERENCES	1.	10 CFR 50, Appendix J, Option B.
	2.	FSAR Chapter 15.
	3.	FSAR Section 6.2.
	4.	ASME Code, Section III, Division 2, 2004.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

One personnel air lock and one emergency air lock provide access to the Containment. Each air lock is nominally a right circular cylinder, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 2). This leakage rate is defined in

BASES

APPLICABLE SAFETY ANALYSES (continued)

10 CFR 50, Appendix J, Option B (Ref. 1), as L_a = 0.25% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 55 psig following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during movement of recently irradiated fuel assemblies within containment are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel

ACTIONS (continued)

side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

<u>A.1, A.2, and A.3</u>

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

BASES

ACTIONS (continued)

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

<u>B.1, B.2, and B.3</u>

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable.

BASES

ACTIONS (continued)

With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

BASES

ACTIONS (continued)

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment

BASES

SURVEILLANCE REQUIREMENTS (continued)

OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 months Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 months Frequency for the interlock is justified based on generic operating experience. The 24 months Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the airlock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.

2. FSAR Section 6.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Isolation stage 1 (CI-1) occurs upon Containment pressure > MAX1p, or receipt of a Safety Injection System (SIS) signal. The CI-1 signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Isolation stage 2 (CI-2) occurs upon Containment pressure > MAX2p. The CI-2 signal isolates the remaining process lines, except systems required for accident mitigation.

As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

BASES

BACKGROUND (continued)

Full Flow Purge Sub-system (39 inch purge valves)

The Full Flow Purge portion of the Containment Ventilation System operates to supply outside air into the containment for ventilation and cooling or heating during a unit outage. The supply and exhaust lines each contain two isolation valves. The 39 inch full flow purge valves are qualified for automatic closure from their open position under DBA conditions. The Full Flow Purge Sub-system is not needed in MODES 1, 2, 3, and 4 and the 39 inch full flow purge valves are maintained closed to ensure the containment boundary is maintained.

Low Flow Purge Sub-system (20 inch purge valves)

The Low Flow Purge Sub-system operates to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access and

b. Equalize internal and external pressures.

Since the valves used in the Low Flow Purge Sub-system are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), fuel handling and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the 39 inch full flow purge valves are closed at the start of a LOCA or rod ejection but not for a fuel handling accident. The DBA analysis assumes that, within

BASES

APPLICABLE SAFETY ANALYSES (continued)

60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the design of the full flow purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves are pneumatically operated spring closed valves that will fail on the loss of air.

The full flow purge valves are designed to close in the environment following a LOCA or MSLB. However, the DBA dose analysis assumes that each full flow purge line is isolated during MODES 1, 2, 3, and 4.

The low flow purge valves may be opened during normal operation. In this case, the single failure criterion remains applicable to the low flow purge valves due to failure in the control circuit associated with each valve. The system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 39 inch full flow purge valves must be maintained sealed closed. The valves covered by this LCO are listed along with their associated stroke times in FSAR Section 6.2.4 (Ref. 2).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 2.

BASES

LCO (continued)

Purge valves with resilient seals must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during movement of recently irradiated fuel assemblies within containment are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for 39 inch full flow purge flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the full flow purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. A single full flow purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

BASES

ACTIONS (continued)

In the event the isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for purge valve leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This

Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

BASES

ACTIONS (continued)

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

B.1

With two or more containment isolation valves in one or more penetration flow paths inoperable, except for purge valve leakage not within limit, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

BASES

ACTIONS (continued)

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two or more containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Ref. 3. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices

BASES

ACTIONS (continued)

to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1, D.2, and D.3

In the event one or more full flow purge valves in one or more penetration flow paths are not within the full flow purge valve leakage limits, purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A full flow purge valve with resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.6. The specified Completion Time is reasonable, considering that one full flow purge valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification that those isolation devices outside containment capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the full flow purge valve with resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.3.6 must be performed at least once every 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the full flow purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.6, 184 days, is based on an NRC

BASES

ACTIONS (continued)

initiative, Generic Issue B-20 (Ref. 4). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

Required Action D.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

E.1 and E.2

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 39 inch full flow purge valve is required to be verified closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a full flow purge valve. The full flow purge valves are designed to close in the environment following a LOCA. However, the DBA dose analysis assumes that each full flow purge line is isolated during MODES 1, 2, 3, and 4.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2

This SR ensures that the low flow purge valves are closed as required or, if open, open for an allowable reason. If a low flow purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the low flow purge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The low flow purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.6

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 4).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES:

1. FSAR Chapter 15.
2. FSAR Section 6.2.
3. NUREG 0800, Section 6.2.4.
4. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."
5. Generic Issue B-24, "Seismic Qualification of Electrical and Mechanical Components."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND	The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or main steam line break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of transients which result in a negative pressure. Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values and may result in leakage greater than assumed in the accident analysis.
APPLICABLE SAFETY ANALYSES	Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and MSLB, which are analyzed using computer pressure transients. The worst case MSLB generates larger mass and energy release than the worst case LOCA. Thus, the MSLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1). The initial pressure condition used in the containment analysis was 15.96 psia (1.26 psig). This resulted in a maximum peak pressure from a LOCA of 52.0 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, results from the limiting LOCA. P_a is conservatively set at 55 psig. The maximum containment pressure resulting from the worst case LOCA, 52.0 psig, does not exceed the containment design pressure, 62 psig. The containment was also designed for an external pressure load equivalent to -3.0 psig An inadvertent actuation of the Severe Accident Heat Removal System is not considered a credible event for the U.S. EPR since it is manually actuated for beyond design basis events only. An evaluation of a Containment cooldown event determined a worse case pressure of -2.9 psig. Containment pressure satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO	Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following negative pressure transients.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.
ACTIONS	<u>A.1</u> When containment pressure is not within the limit of the LCO, it must be restored to within this limit within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour. <u>B.1 and B.2</u> If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.4.1 Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.
REFERENCES	1. FSAR Section 6.2. 2. 10 CFR 50, Appendix K.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND	The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or main steam line break (MSLB). The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Low Head Safety Injection System (LHSI) during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.
APPLICABLE SAFETY ANALYSES	Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1). The limiting DBAs considered relative to containment OPERABILITY are the LOCA and MSLB. The DBA LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to ESF Systems, assuming a worst case single active failure as identified in Reference 5.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting DBA for the maximum peak containment air temperature is an MSLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 131°F. This resulted in a maximum containment air temperature of 428°F. The containment has been qualified for this temperature (Ref. 2).

The temperature limit is used to establish the environmental qualification operating envelope for containment. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Refs. 3 and 4).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is an MSLB. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature and that required safety related equipment will continue to perform its function.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

BASES

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, a weighted average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. FSAR Section 6.2.
2. FSAR Section 3.8.
3. 10 CFR 50.49.
4. FSAR Section 3.11.
5. FSAR Section 15.0.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Shield Building

BASES

BACKGROUND	The shield building is a concrete structure that surrounds the Containment Building. Between the Containment Building and the shield building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the Containment Building. The Annulus Ventilation System (AVS) establishes a negative pressure in the annulus between the shield building and the containment building. Filters in the system then control the release of radioactive contaminants to the environment. The shield building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the AVS.
APPLICABLE SAFETY ANALYSES	The design basis for shield building OPERABILITY is a LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses. The shield building satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).
LCO	Shield building OPERABILITY must be maintained to ensure proper operation of the AVS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses. The LCO is modified by a Note allowing the shield building boundary to be opened intermittently under administrative controls. This Note only applies to openings in the shield building boundary that can be rapidly restored to the design conditions, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be accomplished by procedures, and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

BASES

APPLICABILITY

Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the containment following a LOCA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when a main steam line break, LOCA, or rod ejection accident could release radioactive material to the containment atmosphere.

In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, shield building OPERABILITY is not required in MODE 5 or 6.

ACTIONS

A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period.

B.1 and B.2

If the shield building cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying that shield building annulus negative pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Maintaining shield building OPERABILITY requires verifying each access opening door is closed. However, all shield building access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.6.6.3 and 3.6.6.4

The Annulus Ventilation System (AVS) exhausts the annulus atmosphere to the environment through appropriate treatment equipment. Each safety AVS train is designed to draw down the annulus to a negative pressure of ≥ 0.25 inches of water gauge (wg) in ≤ 305 seconds and maintain the annulus at a negative pressure ≥ 0.25 inches wg. To ensure that all fission products released to the annulus are treated, SR 3.6.6.3 and SR 3.6.6.4 verify that a pressure in the annulus that is less than the lowest postulated pressure external to the shield building boundary can be established and maintained. When the AVS System is operating as designed, the establishment and maintenance of annulus pressure cannot be accomplished if the shield building boundary is not intact. Establishment of this pressure is confirmed by SR 3.6.6.3, which demonstrates that the annulus can be drawn down to a negative pressure ≥ 0.25 inches wg using one AVS train. SR 3.6.6.4 demonstrates that the annulus can be maintained at a negative pressure ≥ 0.25 inches wg using one AVS train at a flow rate ≤ 1320 cfm. The primary purpose of these SRs is to ensure annulus boundary integrity. The secondary purpose of these SRs is to ensure that the AVS train being tested functions as designed. There is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the AVS System. These SRs need not be performed with each safety AVS train. The AVS train used for these Surveillances is staggered to ensure that in addition to the requirements of LCO 3.6.7, either safety AVS train will perform this test. The inoperability of the AVS System does not necessarily constitute a failure of these Surveillances relative to the shield building OPERABILITY. Operating experience has shown the shield building boundary usually passes these Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Annulus Ventilation System (AVS)

BASES

BACKGROUND

The AVS is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials that leak from the Containment Building into the shield building (secondary containment) following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The Containment Building is surrounded by a secondary containment called the shield building, which is a concrete structure. Between the Containment Building and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the Containment Building.

The AVS maintains a negative pressure in the annulus between the shield building and the Containment Building during operation. Filters in the system control the release of radioactive contaminants to the environment. Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the AVS. The AVS is designed to permit appropriate periodic pressure and functional testing to assure component integrity, OPERABILITY of active components, and operational performance of the system as required by GDC-43 "Testing of Containment Atmosphere Cleanup Systems" (Ref 4).

The AVS consists of one normal operation filtration train (non-safety related), and two independent and redundant accident filtration trains (safety related). The normal filtration train operates during normal plant operation, including cold shutdown and outages. During normal plant operation, the accident filtration trains are not required to be in operation, however they are both available for back-up if the normal filtration train is not able to maintain sufficient negative pressure in the annulus.

During normal operation, the conditioned air is drawn from the Nuclear Auxiliary Building Ventilation supply shaft to the bottom of annulus through a fire damper, manual regulated control damper, and two motor operated isolation dampers. The exhaust air is drawn through a vent at the top of annulus through two motor operated isolation dampers and fire dampers to the Nuclear Auxiliary Building Ventilation system exhaust fans via air shaft cell 3. See FSAR Section 9.4.3 (Ref. 5). The exhaust air from cell 3 is filtered by the pre-filter and HEPA filter and then discharged through the vent stack. The annulus air inlet and exhaust motor operated isolation dampers of the normal filtration train are the only components which are safety related.

BASES

BACKGROUND (continued)

The AVS accident filtration trains are used during a design basis event to contain leaks from the primary containment by maintaining a negative pressure in the annulus. During a design basis event, the annulus air is filtered before releasing to the environment. There are two independent 100% accident trains. Each train consists of an upstream air-tight motor controlled damper, electrical heater, pre-filter, upstream HEPA filter, an activated charcoal adsorber for removal of radio-iodines, downstream HEPA filter, downstream air-tight motor controlled damper, fan, and back-draft damper. The downstream bank of HEPA filters following the charcoal adsorber collects carbon particles and provide backup in case of failure of the main HEPA filter bank. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a containment isolation signal. The system is described in Reference 2.

The prefilters remove large particles in the air, and the moisture separators remove entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters may be included to reduce the relative humidity of the airstream on systems that operate in high humidity. Continuous operation of each train, for at least 10 hours per month, with heaters on, reduces moisture buildup on their HEPA filters and adsorbers.

During normal operation, the AVS normal operation filtration train (non-safety related) maintains a negative pressure in the annulus and processes the air through HEPA filters.

The AVS accident filtration train reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the AVS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The AVS design basis is to mitigate the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) assumes that only one train of the AVS is OPERABLE due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA. For all events analyzed, the AVS is assumed to be automatically initiated to reduce via filtration and adsorption, the radioactive material released to the environment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The modeled AVS actuation in the safety analyses is based upon a worst case response time following a containment isolation initiated at the limiting setpoint. The total response time, from exceeding the signal setpoint to attaining the negative pressure of 0.25 inches wg in the shield building, is 305 seconds. This response time is composed of signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting.

The AVS satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

In the event of a DBA, one AVS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two safety related trains of the AVS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the AVS is not required to be OPERABLE (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

ACTIONS

A.1

With one AVS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant AVS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

BASES

ACTIONS (continued)

B.1 and B.2

If the AVS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Operating each AVS train for ≥ 10 continuous hours ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating units indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.7.2

This SR verifies that the required AVS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.7.3

The automatic startup ensures that each AVS train responds properly. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the

BASES

SURVEILLANCE REQUIREMENTS (continued)

potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the AVS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.7.1.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. FSAR Section 6.2.
3. FSAR Chapter 15.
4. 10 CFR 50, Appendix A, GDC 43.
5. FSAR Section 9.4.3.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 pH Adjustment

BASES

BACKGROUND

The U.S. EPR design includes pH adjustment baskets that provide adjustment of the pH of the water in the containment following an accident where the containment floods.

Following an accident with a large release of radioactivity, the containment pH is automatically adjusted to greater than or equal to 7.0,to enhance iodine retention in the containment water (Ref. 1). Chemical addition is necessary to counter the affects of the boric acid contained in the safety injection supplies and acids produced in the post-Loss of Coolant Accident (LOCA) environment (nitric acid from the irradiation of water and air and hydrochloric acid from irradiation and pyrolysis of electric cable insulation). The desired pH values significantly reduce formation of elemental iodine in the containment water, which reduces the production of organic iodine and the total airborne iodine in the containment. This pH adjustment is also provided to prevent stress corrosion cracking of safety related containment components during long-term cooling.

Dodecahydrate trisodium phosphate (TSP) contained in four baskets provides a passive means of pH control for such accidents. The baskets are made of stainless steel with a mesh front that readily permits contact with water. These baskets are located inside containment in a trough in the heavy floor adjacent to the four IRWST strainer openings. Recirculation of water inside the containment, following a LOCA, is driven by the core decay heat and provides mixing to achieve a uniform solution pH. The dodecahydrate form of TSP (Na_3PO_4-$12H_2O$) is initially loaded into the baskets because it is hydrated and will undergo less physical and chemical change than would anhydrous TSP as a result of the humidity inside containment (Reference 1).

APPLICABLE SAFETY ANALYSIS

The LOCA radiological consequences analyses takes credit for iodine retention in the IRWST. In the event of a Design Basis Accident (DBA), iodine may be released from the fuel into the containment. To limit this iodine release from containment, the pH of the water in the IRWST is adjusted by the addition of TSP. Adjusting the IRWST to neutral or alkaline pH (pH $\geq$ 7.0) will augment the retention of the iodine, and thus reduce the iodine available to leak to the environment.

The pH adjustment satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO	The TSP is required to adjust the pH of the recirculation water to > 7.0 after a LOCA. A pH > 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH > 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components. A required volume is specified instead of mass because it is not feasible to weigh the TSP in the containment. The minimum required volume is based on the manufactured density of TSP (58 lb/ft^3). This is conservative because the density of TSP may increase after installation due to compaction.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause release of radioactive iodine to containment requiring pH adjustment. The pH adjustment baskets assist in reducing the airborne iodine fission product inventory available for release to the environment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, pH adjustment is not required to be OPERABLE in MODES 5 and 6.
ACTIONS	<u>A.1</u> If the TSP volume in the baskets is not within limits, the iodine retention may be less than that assumed in the accident analysis for the limiting DBA. Due to the very low probability that the volume of TSP may change, the variations are expected to be minor such that the required capability is substantially available. The 72 hour Completion Time for restoration to within limits is consistent with times applied to minor degradations of ECCS parameters. <u>B.1 and B.2</u> If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.8.1 The minimum amount of TSP is 211 ft^3. This volume is based on providing sufficient TSP to buffer the post-accident containment water to a minimum pH of 7.0. Additionally, the TSP volume is based on treating the maximum volume of post-accident water (628,320 gallons) containing the maximum amount of boron (1800 ppm) as well as other sources of acid. The minimum required mass of TSP is 12,200 pounds. The minimum required volume of TSP is based on this minimum required mass of TSP, the minimum density of TSP plus margin to account for degradation of TSP during plant operation. The minimum TSP density is based on the manufactured density, since the density may increase and the volume decrease, during plant operation, due to agglomeration from humidity inside the containment. The minimum required TSP volume also has approximately 10% margin to account for degradation of TSP during plant operation. The periodic verification is required every 24 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 24 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the Containment Building.
REFERENCES	1. FSAR Section 6.3.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The MSSVs provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurization of the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the condenser, is not available. This is done in conjunction with the Emergency Feedwater System (EFW) providing cooling water from the EFW Storage Pools.

The MSSVs are spring-loaded safety valves. Two MSSVs are located on each Main Steam Line, outside containment, upstream of the main steam isolation valves and downstream of the Main Steam Relief Train (MSRT), as described in FSAR Section 10.3 (Ref. 1).

The MSSVs along with the MSRTs provide overpressure protection of the main steam piping and steam generators. Together, the MSSVs and MSRTs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV setpoints and capacities are such that with consideration of reactor trip, the MSSVs alone will prevent main steam pressure from rising above 110% of the steam generator design pressure upon full loss of load.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure during an anticipated operational occurrence (AOO) or postulated accident.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in FSAR Section 15.2 (Ref. 3). Of these, the closure of a single main steam isolation valve without main steam bypass or partial trip function is the limiting AOO. Closure of a single MSIV results in a smaller isolated volume on the secondary side, therefore this event is more limiting than a turbine trip event for secondary system over pressure.

The safety analysis demonstrates that the transient response for MSSV closure occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSRTs and MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. These events are bounded by the MSSV closure event.

The safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The accident analysis requires that the two MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that the two MSSVs per steam generator be OPERABLE in compliance with Reference 2 and the postulated accident analysis.

The OPERABILITY of the MSSVs are defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and be closed or reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their design safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, two MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSRTs or MSSVs to be OPERABLE in these MODES.

BASES

ACTIONS

A.1 and A.2

With one required MSSV inoperable, the associated MSRT is verified OPERABLE and action must be taken to restore the valve to OPERABLE status within 30 days. Verification of MSRT OPERABILITY is performed as an administrative check by examining logs or other information to determine if an MSRT is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the MSRT. If the OPERABILITY of the associated MSRT cannot be verified, however, Condition C must be immediately entered.

The 30 day Completion Time considers the following:

a. With one MSSV inoperable, the resulting relief capacity of the affected SG is 75% (taking into account the MSRT) of the full load steam generation of the assigned steam generator, which is greater than the 50% relief capacity considered in the safety analysis.

b. The remaining OPERABLE overpressure protection devices are sufficient for heat removal for the long term phase.

The 30 day Completion Time is considered reasonable for restoring the inoperable components to OPERABLE status.

B.1

With two MSSVs inoperable, actions must be taken to restore the inoperable MSSVs to OPERABLE status within 7 days.

This Completion Time is applicable because:

a. With two MSSVs inoperable on the same SG, the resulting relief capacity of the affected SG is 50% (taking into account the MSRT) of the full load steam generation per SG, which is equal to 50% relief capacity considered in the safety analysis.

b. With one MSSV inoperable on one SG and one MSSV inoperable on another SG, the resulting relief capacity for each SG is 75% of the full load steam generation per SG. This combination of inoperabilities is different from the one in the safety analysis. However, the total relief capacity of the four SGs is 350% of the full load steam generation per SG, which is the exact relief capacity considered in the safety analysis.

c. The remaining OPERABLE overpressure protection devices (MSRTs) are sufficient for heat removal in the long term phase.

BASES

ACTIONS (continued)

The 7 day Completion Time is considered reasonable based on operating experience to accomplish the Required Action in an orderly manner without challenging unit systems.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met or if three or more MSSVs are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code (Ref. 4) requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5).

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. The SR allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

BASES

REFERENCES

1. FSAR Section 10.3.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. FSAR Section 15.2.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. ANSI/ASME OM-1-1987.
6. FSAR Section 15.4.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located on each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and main steam relief train (MSRT), to prevent MSSV and MSRT isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, turbine bypass, and other auxiliary steam supplies from the steam generators.

The MSIVs are controlled by two redundant and parallel control lines. Each control line is composed of:

a. Two fast closure pilot valves in series actuating a common fast closure distributor; and

b. An exercise pilot valve actuating an exercise distributor.

The arrangement of pilot valves prevents a failure in any pilot valve to cause either a spurious closing (two pilot valves in series) or a failure to close (two manifolds in parallel). The MSIVs fail safe position is closed on loss of control or power supply. The pilot valves are de-energized to close the MSIVs.

The MSIVs are closed under faulted conditions by the Protection System. The MSIVs can also be closed manually. The MSIVs fail closed on loss of control or actuation power.

A description of the MSIVs is found in FSAR Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the main steam line break (MSLB) inside containment, discussed in FSAR Section 6.2 (Ref. 2). It is also affected by the accident analysis of the MSLB and feedwater line break events presented in FSAR Chapter 15 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting case for the containment analysis is the MSLB inside containment, with offsite power available, and failure of the MSIV on the affected steam generator to close. At lower power levels, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIVs contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power.

The accident analysis compares several different MSLB events against different acceptance criteria. The double-ended guillotine break of a main steam line in the valve compartment in the Safeguards Building upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The MSLB outside containment, upstream of an MSIV at hot zero power is the limiting case for a post trip return to power. The analysis includes a spectrum of break sizes, scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. The worse case single failure is a main steam relief control valve associated with one of the unaffected steam generators failed in the fully open position (Ref. 3).

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

c. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

d. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, two control lines per MSIV are OPERABLE, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 50.34 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated, when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, normally most of the MSIVs are closed and the steam generator energy is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

BASES

ACTIONS

A.1

With only one control line of one or more MSIVs inoperable in MODE 1, the affected MSIV (s) can still be closed by the other control line, however actions must be taken to restore the inoperable control line(s) to OPERABLE status in 72 hours. The 72 hour Completion Time is reasonable considering the MSIV would be closed by the OPERABLE control line in the event of an accident.

B.1

With one MSIV inoperable due to the inoperability of both control lines or reasons other than Condition A, the MSIV must be restored to OPERABLE status within 8 hours. Otherwise the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Times are reasonable based on operating experience to reach MODE 2 and to close the MSIV(s) in an orderly manner without challenging unit systems.

C.1

If Required Action A.1 or B.1 cannot be met within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition D would be entered. The completion times are reasonable based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems.

D.1 and D.2

Condition D is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is reasonable based on operating experience.

BASES

ACTIONS (continued)

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

E.1 and E.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that each MSIV and its pilot valves are OPERABLE, i.e. that it can be closed on demand. The test is performed one valve at a time using one control line only, in MODES 1 and 2, under stable plant conditions. The Surveillance Frequency of 31 days is consistent with operating experience of similar MSIVs on existing plants.

SR 3.7.2.2

This SR verifies freedom of movement of the valve stem and disk by partial valve closure and re-opening. The MSIV design allows for this test during power operation without impairing power generation and without risk of full valve closure. The Surveillance Frequency of 92 days is consistent with operating experience of similar MSIVs on existing plants.

The Frequency is in accordance with the Inservice Testing Program and is in accordance with the ASME Code (Ref. 5). This SR is modified by a Note that limits this surveillance to MODES 1 and 2.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.3

This SR verifies that MSIV closure time is within the limit assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with an owner controlled document controlled under 10 CFR 50.59 and the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs can not be full stroke tested when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program. This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.4

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MSIV testing is every 24 months on a STAGGERED TEST BASIS for each control line. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint. This SR is modified by a NOTE that requires the performance of this surveillance prior to entry into MODE 2.

REFERENCES

1. FSAR Section 10.3.
2. FSAR Section 6.2.
3. FSAR Chapter 15.
4. 10 CFR 50.34.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater (MFW) Valves

BASES

BACKGROUND

On each of the four steam generators (SGs), the Main Feedwater valves (MFW Full Load Isolation Valves (MFWFLIVs), MFW Full Load Control Valves (MFWFLCVs), MFW Low Load Isolation Valves (MFWLLIVs), FW Low Load Control Valves (MFWLLCVs), MFW Very Low Load Control Valves (MFWVLLCVs), and MFW Main Isolation Valves (MFWMIVs)) are located in valve stations, physically separated from each other and from other systems. Within these valve compartments, the MFW lines are arranged in three trains, one Very Low Load, one Low Load, and one Full Load train. The full load flow path for each steam generator includes one MFWFLCV, one MFWFLIV, and the MFWMIV. The low load flow path for each steam generator includes one MWFLLCV, one MFWLLIV, and the MFWMIV. The very low load flow path for each steam generator includes one MFWVLLCV, one MFWLLIV, and the MFWMIV. Each of these trains can be isolated redundantly by one isolation valve, one control valve, or the MFWMIV. The Low Load isolation valve allows isolation of the Low Load and the Very Low Load train at the same time.

The closure of these valves allows limiting the filling of the steam generators in case of a too high feedwater flowrate which could impair the functioning of the safety valves of the Main Steam System.

In the event of a secondary side pipe rupture inside containment, the valves also limit the quantity of high energy fluid that enters containment through the break and provide a pressure boundary for the controlled addition of Emergency Feedwater (EFW) to the intact loops. They also reduce the cooldown effects in case of Main Steam Line Breaks (MSLBs) or in case of excessive increase in feedwater flowrate caused by a feedwater system malfunction.

A MFW Isolation valve outside containment and a MFW check valve inside containment provide the containment isolation function.

The MFWFLIVs and MFWFLCVs close on a reactor trip. The low and low-low range control and isolation valves close in response to steam generator level as described in Reference 1. The MFWMIV closes on a containment isolation signal. The MFW valves may also be actuated manually.

A description of the MFW valves is found in FSAR Section 10.4.7 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES	The Full Load Line must be isolated on each of the four SGs by redundant means in case of reactor trip or a high SG level signal. The Low and Very Low Load line of the affected SG must be isolated in case of high level, low pressure or a high pressure drop signal coming from the SG. These actions are needed to mitigate the following accidents: MSLB; Feedwater Line Break (FWLB); Steam Generator Tube Rupture (SGTR); or Feedwater Malfunction. The failure of these respective valves to close could lead to an overcooling event causing re-criticality (in case of MSLB or feedwater malfunction), to increase the mass and energy releases inside containment (in case of MSLB or FWLB) or to fill the steam lines with feedwater (in case of SGTR or feedwater malfunction). The MFW valves close on reactor trip and feedwater isolation signals as described in detail in Ref. 1. Each flow path has three isolation or control valves in series in addition to a check valve located inside Containment. The MFW valves satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).
LCO	This LCO ensures that the MFW isolation and control valves will reduce or isolate MFW flow to the steam generators, as required, following an excessive feedwater flow accident, a FWLB, an SLB, or an SGTR. It also ensures that the MFWMIV provides isolation for events requiring containment isolation. This LCO requires that four MFWFLIVs, four MFWFLCVs, four MFWLLIVs, four MFWLLCVs, four MFWVLLCVs, and four MWFMIVs be OPERABLE. The MFWFLIVs, MFWFLCVs, MFWLLIVs, MFWLLCVs, MFWVLLCVs, and MFWMIVs are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal. Failure to meet the LCO requirements can result in additional mass and energy being released to containment, the introduction of water into the main steam lines, or an overcooling of the primary circuit depending on the accident considered.
APPLICABILITY	The feedwater isolation and control valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and in the steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODE 1 and in MODES 2 and 3 except when closed and de-activated, the full-load, low load, and very low load isolation and control valves are required to be OPERABLE to limit the amount of water in the steam generator, to limit the overcooling of the primary circuit, or to limit the amount of water that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated, they are already performing their safety function.

BASES

APPLICABILITY (continued)

In MODES 4, 5, and 6, steam generator energy is low and all MFW valves are normally closed since MFW is not required.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each flow path.

A.1

With one valve in the full load flow path inoperable, action must be taken to restore the affected valve to OPERABLE status within 7 days. The 7 day Completion Time takes into account the redundancy afforded by the associated MFWFLCV, MFWFLIV, and MFWMIV; and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 7 day Completion Time is reasonable, based on operating experience.

B.1

With two valves in the full load flow path inoperable, action must be taken to restore the affected valves to OPERABLE status within 72 hours. The 72 hour Completion Time takes into account the redundancy afforded by the associated MFWFLCV, MFWFLIV, and MFWMIV; and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 72 hour Completion Time is reasonable, based on operating experience.

C.1

With three valves in the full load flow path inoperable, action must be taken to restore the affected valves to OPERABLE status within 8 hours. The 8 hour Completion Time takes into account the redundancy afforded by the redundant actuation trains on MFW full load flow path valves and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 8 hour Completion Time is reasonable based on operating experience.

BASES

ACTIONS (continued)

D.1 and D.2

With one or more MFWLLIVs, MFWLLCVs, or MFWVLLCVs in the low load or very low load flow path inoperable, action must be taken to restore the affected valve to OPERABLE status within 8 hours or isolate the flow path. When the valves are closed, they are performing their required safety function

Inoperable MFW low load and very low load flow path valves that are closed as a result of this Required Action, must be verified on a periodic basis that they are closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

E.1 and E.2

If the MFWs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFWFLIV, MFWFLCV, MFWLLIV, MFWLLCV, MFWVLLCV, and MFWMIV is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time in accordance with an owner controlled document controlled under 10 CFR 50.59 and the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.2

This SR verifies that each valve can close on an actual or simulated actuation signal. This Surveillance is normally performed during shutdown or upon returning the plant to operation following a refueling outage.

The Frequency for this SR is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 10.4.7.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.4 Main Steam Relief Trains (MSRTs)

BASES

BACKGROUND

The MSRTs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the condenser not be available. This is done in conjunction with the Feedwater or Emergency Feedwater System. The MSRT valves also provide secondary overpressure protection.

One MSRT is provided for each steam generator, outside containment, upstream of the Main Steam Safety Valves and the main steam isolation valves. Each MSRT consists of one main steam relief control valve (MSRCV) located downstream of one main steam relief isolation valve (MSRIV).

The main steam relief control valves are motorized control valves, normally open, which allow control of the steam generator steam pressure, and consequently control of the cooldown rate. The MSRCVs provide a means of controlling MSRT steam flow to prevent overcooling the RCS. The MSRCVs allow mitigation of the effects of a stuck open MSRIV. The MSRCVs are automatically positioned based on thermal power.

The main steam relief isolation valves are angle globe valves with a motive steam-operated piston actuator, operated by two parallel sets of two pilot valves in series. The arrangement of pilot valves prevents a failure in any pilot valve from causing either a spurious opening (two pilot valves in series) or a failure to open (two sets of pilot valves in parallel). The MSRIVs close (fail safe position) on loss of power supply or on loss of Instrumentation and Control. The pilot valves must be energized to open the associated MSRIV.

The MSRIVs are normally closed, with the pilot valves kept closed (de-energized). The valves open automatically and quickly on demand from the Protection System.

A description of the main steam relief control valves and of the main steam relief isolation valves is found in FSAR Section 10.3 (Ref. 1)

Each MSRT minimum required capacity is 50% of the full steam generation of the assigned steam generator (for a design core power level of 4590 MWth), at a design pressure of 1,435 psig, thus limiting the system pressure to ≤ 110% of the steam generator design pressure, in order to meet the requirements of the ASME Code, Section III (Ref. 2). The minimum required capacity, combined with the MSSV capacity, provides 100% flow relief at steam generator design pressure per SG.

BASES

BACKGROUND (continued)

Each MSRT maximum capacity is limited to 50% of the full load steam generation of its assigned steam generator (at a design core power of 4590 MWth), at design pressure of 1,435 psig, thus limiting the consequences of MSRIV spurious opening with regards to reactor coolant system overcooling and reactivity control.

The MSRTs are actuated automatically by the Protection System, but can be controlled manually by the operator.

APPLICABLE SAFETY ANALYSES

The design basis of the main steam relief valves is established by the capability to remove residual heat in a controlled manner and to cool down the plant to RHRS entry conditions at various rates (normal cooldown at 90°F/h, partial cooldown at 180°F/h). The design rate of partial cooldown is applicable to events with two steam generators OPERABLE, each with one MSRT OPERABLE.

The MSRTs residual heat removal function in the safety analysis of Reference 3 is required:

a. To perform residual heat removal at the controlled rate following Condition II, III, and IV events with the condenser inoperable;

b. To perform cooldown to RHR entry conditions, following Condition II, III, and IV events with the condenser inoperable; and

c. To perform Partial Cooldown of the unit at a rate of 180°F/h from MODE 3 to 870 psia to allow Medium Head Safety Injection (MHSI) into the Reactor Coolant System in the event of a Loss of Coolant Accident (LOCA) or Steam Generator Tube Rupture (SGTR).

The Main Steam Relief Trains do not directly participate in the reactivity control function. Nevertheless, reactivity control is supported by isolating a spuriously open MSRT to limit RCS cooling. Excessive increase in steam flow causes overcooling of the reactor coolant and thus reactivity feedback to the core.

In case of a SGTR, the MSRT participates in the confinement of radioactive material. In the SGTR mitigation process, an increase in the MSRV setpoint of the affected SG over the MHSI delivery pressure enables termination of the leak flow. It also prevents overfilling of the affected SG. In the event that the condenser in inoperable, the MSRT challenge avoids response of the associated MSSVs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MSRT participates in the limitation of SG pressure increase in the most limiting Category II overpressure transient, inadvertent closure of one MSIV, thus limiting the SG pressure peak to less than 110% of the design pressure, without MSSV challenge. In other overpressure transients (Category III and IV), the MSRT participates in the limitation of the pressure peak in conjunction with the associated MSSV (see B 3.7.1). For the most limiting overpressure transient of Category 4 (i.e. loss of secondary side heat sink at full power without reactor trip), the inoperability of one MSRT is considered in the safety analysis and all other overpressure protection devices (other MSRTs and all MSSVs) are challenged and thus must be operable.

In all analyzed events, two steam generators and consequently both their MSRTs are required for residual heat removal and plant cooldown, considering a single failure of one MSRT and preventive maintenance performed on either electrical division or emergency diesel at the moment of the accident with assumed Loss of Offsite Power.

In events analyzed in Reference 3, the MSRT ensures residual heat removal by performing either Partial Cooldown or Fast Cooldown, either by automatic or manual action, depending on the event.

The MSRV position is automatically controlled by the Protection System as a function of power level, provided that the MSRIV is closed. The MSRV position is such that:

a. Consequences of a spurious MSRT event are limited with regards to the Reactor Coolant System; and

b. Mitigation of overpressure transients is ensured.

If the MSRIV opens, the MSRV is automatically switched into SG pressure control mode.

The MSRT valves satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Four main steam relief trains (MSRIVs and MSRCVs) are required to be OPERABLE so as to ensure residual heat removal by a minimum of two steam generators even in the case of preventative maintenance and a single failure affecting the other two steam generators and connected cooling systems (e.g., Emergency Feedwater System, MSRT).

Isolation capability is also required on the four MSRTs, since any steam generator can be affected by a spuriously opened MSRIV or by an SGTR.

BASES

LCO (continued)

Failure to meet the LCO either results in the inability to perform residual heat removal and plant cooldown following an event with an inoperable condenser, or in the inability to isolate a SG affected by an SGTR or a spurious MSRIV opening.

A main steam relief control valve is considered OPERABLE when it is capable of full opening and closing and when it is capable of providing controlled relief of the main steam flow, with support of related I&C systems.

A main steam relief isolation valve is considered OPERABLE when it is capable of opening and when it is capable of re-closure after challenge.

APPLICABILITY

In MODES 1, 2, and 3, the MSRTs are required to be OPERABLE to provide a decay heat removal path in conjunction with the Emergency Feedwater System.

In MODE 4, 5, or 6, decay heat removal is provided by the Low Head Safety Injection System.

ACTIONS

A.1

With one control line inoperable for opening in one or more MSRIVs (i.e. one pilot valve is blocked closed), the affected MSRIVs are still OPERABLE, however the control line(s) must be restored to OPERABLE status in 30 days. This completion time is based on the following:

a. Redundancy for MSRIV opening is provided by the second control line.

b. In case of an event with loss of the condenser and assuming a single failure on the second control line of one MSRIV to open, the residual heat removal can still be ensured by the other MSRTs.

c. In case of an overpressure event and assuming a single failure of the second control line of one MSRIV to open which leads to failure to open of the associated MSRIV, the redundancy provided by the two associated OPERABLE MSSVs ensure the pressure limitation in the affected SG.

BASES

ACTIONS (continued)

In case one pilot valve is blocked open in at least one control line of one or more MSRIVs, the isolation function of the MSRIV is not assured. The control line(s) must be restored to OPERABLE status in 30 days. This Completion Time is based on redundancy for MSRIV closure provided by the second pilot valve in series and by the MSRCV.

B.1 and B.2

With one or two MSRIVs inoperable for opening (e.g., due to a mechanical failure or due to two pilots in parallel blocked closed), the residual heat removal function and the overpressure protection function of the corresponding MSRT are not assured. The associated MSSV must be verified OPERABLE and the valves must be restored to OPERABLE status in 7 days. Verification of MSSV OPERABILITY is performed as an administrative check by examining logs or other information to determine if an MSSV is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the MSSV. If the OPERABILITY of the associated MSSV cannot be verified, however, Condition C must be immediately entered.

The Completion Time of 7 days is reasonable because:

a. In case of an event with loss of condenser, the two OPERABLE MSRTs are capable of performing the residual heat removal function. However the single failure criterion may not be met.

b. In case of an overpressure event, the redundancy provided by the two OPERABLE MSSVs ensure the limitation of SG pressure (the necessary relief capacity being 50% of full load steam generation per SG). However the single failure criterion is not met.

With one or two MSRCVs inoperable, the residual heat removal function and the overpressure protection of the corresponding MSRT are not assured, as well as the isolation function. The single failure criterion is not fulfilled any more, so the same Completion time of 7 days applies to restore MSRCV(s) to OPERABLE status.

Finally, with one or two MSRIVs inoperable for closing (e.g., blocked open during residual heat removal with MSRTs or due to a failed test), the residual heat removal function is still ensured, but the redundancy for MSRT isolation is lost because it can only be ensured by associated MSRVs. As a result, the same Completion Time of 7 days applies to restore to OPERABLE status.

BASES

ACTIONS (continued)

C.1 and C.2

If required Action A.1 or B.1 cannot be met within the required Completion Times, the unit must be placed in a MODE in which the LCO does not apply, and in which the inoperable MSRCV or MSRIVs can be restored to OPERABLE status. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

With three or more MSRIVs inoperable for opening, or three or more MSRCVs inoperable, the residual heat removal function and the overpressure protection function of the corresponding MSRT are not assured. Only one MSRT remains OPERABLE for this function, which is less than the needed two MSRTs for residual heat removal following analyzed events.

The unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4, without reliance upon steam generators for heat removal, within 12 hours.

The MSRCVs inoperability also affects the MSRT isolation function by loss of redundancy (only MSRIVs can ensure the function).

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies each MSRIV OPERABLITY by opening the valve and then by closing the MSRIV. This SR is performed once every refueling outage on a STAGGERED TEST BASIS for each control line (i.e. twice per MSRIV) in hot shutdown conditions. The frequency is reasonable based on the fact that complete opening of an MSRIV is not possible during power operation and on the operating experience of similar MSRIVs on existing plants.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.2

This SR verifies each MSRCV OPERABILITY by stroking the valve through a full cycle. The test can be performed during power operation under stable conditions without impairing power operation because the MSRIV stays closed during the test. The test can also be performed in hot shutdown conditions before plant shutdown. The frequency of once per cycle is reasonable based on operating experience and on the fact that the MSRCV operates under the control of the Protection System during power operation, which ensures that the valve is not blocked in a specific position.

SR 3.7.4.3

This SR demonstrates that each MSRIV actuates on an actual or simulated steam pressure setpoint signal. The 24 month frequency is based on the need to perform the test during either hot or cold shutdown conditions. The frequency is reasonable based on the fact that opening a MSRIV is not possible during power operation and on operating experience of similar MSRIVs on existing plants.

SR 3.7.4.4 and 3.7.4.5

This SR demonstrates that each MSRCV is automatically positioned based on thermal power and is switched into SG pressure control mode on an actual or simulated MSRIV opening. The test can be performed in hot shutdown conditions before plant shutdown. The frequency of once per cycle is reasonable based on operating experience and on the fact that the MSRCV operates under the control of the Protection System during power operation, which ensures that the valve is not blocked in a specific position.

REFERENCES

1. FSAR Section 10.3.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

3. Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System on low Steam Generator (SG) level or Loss of Offsite Power. The EFW pumps take suction through a common supply header from their respective EFW storage pool (SP) and normally pump to their respective steam generator secondary side via separate and independent connections. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or main steam relief valves (MSRVs) (LCO 3.7.4). If the main condenser is available, steam may be released via the turbine bypass valves.

The EFW System consists of four motor driven EFW pumps and four EFW SPs configured into four separate trains. The inventory of the four EFW SPs is available to all EFW pumps through the common supply header.

The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each EFW pump is powered from an independent Class 1E power source.

The non-safety Startup and Shutdown System (SSS) is used for supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The EFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the EFW System supplies sufficient water to cool the unit to Low Head Safety Injection (LHSI) entry conditions, with steam released through the main steam relief valves.

The EFW System actuates automatically on low steam generator water level signal generated by the Protection System (LCO 3.3.1). The system also actuates on loss of offsite power signal.

The EFW System is discussed in FSAR Section 10.4.9 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The EFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the EFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3%.

In addition, the EFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

There are four EFW trains. Each EFW train has a separate SP. All four EFW SPs, the common supply and discharge headers, and the four injection paths are required to be OPERABLE. One EFW pump train is assumed to be unavailable due to maintenance and a second EFW pump train or its normal injection pathway is assumed to be lost to a single failure. Note, an EFW Pump Train includes the pump, discharge check valve, flow control valve, and piping to the manual isolation valves on the suction and discharge of the pump.

The two remaining EFW trains provide sufficient flow for decay heat removal as required by the accident analysis. For certain sized feedwater line breaks, one of the remaining EFW pumps feeds a faulted steam generator. This pump is re-aligned from the MCR at 30 minutes to feed through the injection pathway associated with the train whose pump is unavailable due to maintenance.

The limiting accident for the EFW System is a Main Feedwater Line Break (MFWLB) with a natural circulation cooldown.

In addition, the minimum available EFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident (LOCA).

The Protection System automatically actuates the EFW pumps and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power.

The EFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(d)(2)(ii) for operation in MODES 1, 2, and 3 and Criterion 4 of 10 CFR 50.36(d)(2)(ii) for operation in MODE 4.

BASES

LCO

This LCO provides assurance that the EFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary.

Four EFW pumps, the common supply and discharge headers, and the four injection paths are required to be OPERABLE to ensure decay heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering each of the pumps from independent emergency buses.

The EFW System is configured into four trains, which share common supply and discharge headers. The EFW System is considered OPERABLE when the components and common flow paths required to provide redundant EFW flow to the steam generators are OPERABLE. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

In MODE 4 with only one EFW pump OPERABLE, operation is allowed to continue because only one EFW pump is required in accordance with the Note that modifies the LCO. Because of the reduced heat removal requirements and the short period of time in MODE 4, one EFW pump is sufficient to remove decay heat. Although not required, the unit may continue to cool down to LHSI entry conditions.

APPLICABILITY

In MODES 1, 2, and 3, the EFW System is required to be OPERABLE in the event that it is called upon to function when MFW and offsite power are lost. In addition, the EFW System is required to supply enough makeup water to replace the secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4 and 5, the EFW System may be used for heat removal via the steam generators.

In MODE 6, the steam generators are not normally used for heat removal, and the EFW System is not required

ACTIONS

A Note prohibits the application of LCO 3.0.4.b for two or more EFW trains inoperable when entering MODE 1. There is an increased risk associated with entering MODE 1 with two or more EFW trains inoperable and the provisions of LCO 3.0.4.b, which allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

BASES

ACTIONS (continued)

A.1

With one EFW train inoperable in MODE 1, 2, or 3, action must be taken to restore OPERABLE status within 120 days. The 120 day Completion Time is reasonable, based on the FSAR Chapter 15 analysis assumption that one EFW train is not available due to maintenance, and the low probability of a postulated accident occurring during this time period.

B.1

With two EFW trains inoperable in MODES 1, 2, or 3, action must be taken to restore at least one inoperable EFW train to OPERABLE status in 72 hours. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of a postulated accident occurring during this time period.

C.1 and C.2

When any Required Action and associated Completion Time cannot be met; or if three EFW trains are inoperable in MODE 1, 2, or 3; or the common injection header; the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

With four EFW trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown and only non-safety means for conducting a cooldown with the SSS. In such a condition, the unit should not be perturbed by any action, including a power change that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one EFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one EFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

BASES

ACTIONS (continued)

E.1

In MODE 4, either the reactor coolant pumps or the LHSI loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the one required EFW pump inoperable, action must be taken to immediately restore an inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW System flow paths provides assurance that the proper flow paths will exist for EFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. For the EFW System, this SR includes the steam generator blowdown isolation valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Each EFW pump suction and supply header isolation valve is required to be locked open at 31 day intervals. This surveillance is designed to ensure that all EFW pumps can the inventory of all EFW pools.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.3

Each EFW discharge header cross-connect valve is required to be cycled in order to assure the capability for any EFW pump to feed any steam generator as assumed in the main feedwater line break (Ref. 3) The Frequency of this SR is in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 2). Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

SR 3.7.5.5

This SR verifies that EFW can be delivered to the appropriate steam generators in the event of any accident or transient that generates a Protection System actuation, by demonstrating that each automatic valve in the flow path actuates to its correct position, each EFW pump starts automatically, and flow rate is controlled within required limits and steam generator level is controlled within limits, on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

SR 3.7.5.6

This SR verifies that the EFW is properly aligned by verifying the flow paths from the supply header to its respective steam generator prior to entering MODE 2 after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of EFW flow paths must be verified before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the SP to the steam generators is properly aligned.

BASES

REFERENCES

1. FSAR Section 10.4.9.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

3. FSAR Section 15.2

B 3.7 PLANT SYSTEMS

B 3.7.6 Emergency Feedwater (EFW) Storage Pools

BASES

BACKGROUND	The EFW pumps take suction through separate suction lines from their respective EFW storage pool (SP) and normally pump to their respective steam generator secondary side via separate and independent connections. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or main steam relief trains (MSRTs) (LCO 3.7.4). If the main condenser is available, steam may be released via the steam bypass valves. The EFW System consists of four motor driven EFW pumps and four EFW SPs configured into four separate trains. The inventory of the four EFW SPs is available to all EFW pumps through the common supply header. Because the SPs are principal components in removing residual heat from the Reactor Coolant System (RCS), they are designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The SPs are designed to Seismic Category I to ensure availability of the feedwater supply. A description of the SPs is found in FSAR Section 10.4.9 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The EFW SPs provide cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in Chapters 6 and 15 (Ref. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally four hours at MODE 3, steaming through the MSSVs and MSRVs followed by a cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate or a lower cooldown rate if offsite power is not available. The limiting accident for the EFW SPs is a Main Feedwater Line Break (MFWLB) with a natural circulation cooldown. The EFW SPs satisfy the requirements of Criterion 2 and 3 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO

To satisfy accident analysis assumptions, the EFW SPs must contain sufficient water to remove decay heat for four hours following a reactor trip from 102% RTP and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the EFW pumps during cooldown or before isolating EFW to a faulted steam generator.

The EFW SP required usable volume of 300,000 gallons is based on a cooldown to RHR entry conditions at 50°F/hour, with all four reactor coolant pumps in service. This basis is established in Reference 1 and exceeds the volume required by the accident analysis.

The OPERABILITY of the EFW SPs is determined by summing the available tank volumes. The volume in an SP is considered usable when it is aligned to the common supply header.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the EFW SPs are required to be OPERABLE to support EFW System operability.

In MODE 5 or 6, the EFW SPs are not required because the EFW System is not required.

ACTIONS

A.1 and A.2

With one EFW SPs inoperable in MODE 1, 2, or 3, or MODE 4, when a steam generator is being relied upon for heat removal, action must be taken to verify the usable volume in the remaining SPs is ≥ 300,000 gal. and to declare the associated EFW train inoperable.

B.1 and B.2

With two or more EFW SPs inoperable or the usable volume of the available SPs is < 300,000 gal., the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4, without reliance on a steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner, and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the EFW Storage Pools contain the required volume of cooling water. The 24 hour Frequency is based on operating experience and are not used by other systems and that the SPs have no other function that to supply water to the EFW trains. Also, the 24 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the SP levels.

SR 3.7.6.2

This SR verifies every 31 days that the EFW supply cross connect valves are locked open. This verification ensures that the usable volume in the SPs are available to all EFW trains through the supply cross connect header and ensures timely discovery if a valve should be not locked open. If an EFW supply cross connect valve is not open, the usable volume of the SP is not available to each of the four EFW trains as assumed in the safety analysis. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned EFW supply cross connect valve is unlikely.

REFERENCES

1. FSAR Section 10.4.9.
2. FSAR Chapter 6.
3. FSAR Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND	The CCW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Service Water (ESW) System and thus to the environment. The CCW System consists of four separate safety classified trains (1, 2, 3 and 4) corresponding to the four layout divisions (1, 2, 3 and 4) and two separate common headers. One of the common headers (common 1 header) is connected normally either to train 1 or to train 2. The other common header (common 2 header) is connected either to train 3 or to train 4. A set of isolation valves per train can separate each train from the common header and either common header is capable of providing safety related cooling of the reactor coolant pump (RCP) thermal barrier cooling common loop. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal from the Protection System, and all nonessential components are isolated. Additional information on the design and operation of the system, along with a list of the components served, is presented in FSAR Section 9.2.2 (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the safety related systems and operational cooling loads to the heat sink via the ESW System. This may be during a normal or post accident cooldown and shutdown.
APPLICABLE SAFETY ANALYSES	The design basis of the CCW System is for two CCW trains to remove the post loss of coolant accident (LOCA) heat load from the In-containment Water Storage Tank (IRWST) by cooling the Low Head Safety Injection System heat exchanger at a maximum CCW temperature of 113°F (Ref. 2). The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA each model the minimum performance of the CCW System, respectively. During a unit cooldown to MODE 5 (T_{cold} < 200°F), a maximum temperature of 113°F is assumed. This maintains the IRWST fluid within acceptable limits following a LOCA as it is supplied to the Reactor Coolant System by the ECCS pumps.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The CCW System also functions to cool the unit from residual heat removal (RHR) entry conditions (T_{cold} < 350°F), to MODE 5 (T_{cold} < 200°F), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR loops operating. Two CCW trains are sufficient to remove decay heat during subsequent operations with T_{cold} < 200°F. This assumes a maximum service water temperature of 95°F occurring simultaneously with the maximum heat loads on the system.

To meet single failure criteria for the RCP thermal barrier cooling function, the load is required to be cooled by a common header which is capable of being connected two OPERABLE CCW trains. A single failure of a train initiates an automatic system response to transfer the common header to the remaining train.

The CCW System satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The CCW System consists of four trains. Four CCW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

A CCW train is considered OPERABLE when:

a. The pump and associated surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

With the exception of the RCP thermal barrier cooling common loop, the isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the Low Head Safety Injection heat exchanger.

BASES

APPLICABILITY (continued)

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A Note has been added to indicated that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable CCW train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

A.1

Required Action A.1 is modified by a Note indicating that the Required Action of A.1 is not applicable if CCW trains are inoperable in both common headers. In this condition, the RCP thermal barrier cooling common loop cannot be aligned to common header capable of being connected to two OPERABLE trains.

If one CCW train is inoperable, action must be taken to align the RCP thermal barrier cooling common loop to a common header capable of being supplied by two OPERABLE CCW trains within 72 hrs. In this condition, the CCW System can perform the RCP thermal barrier cooling function given a single failure. The 72 hour Completion Time is reasonable, based on the low probability of a postulated accident occurring during this period.

A.2

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 120 days. In this condition, the remaining OPERABLE CCW trains are adequate to perform the heat removal function.

B.1

If two CCW trains are inoperable, action must be taken to restore one train to OPERABLE status within 72 hours. In this condition, the remaining OPERABLE CCW trains are adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE trains, and the low probability of a postulated accident occurring during this period.

BASES

ACTIONS (continued)

C.1 and C.2

If a CCW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed

Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components, other than the RCP thermal barrier cooling common loop, may render those components inoperable but does not affect the OPERABILITY of the CCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to

BASES

SURVEILLANCE REQUIREMENTS (continued)

perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 9.2.2.
2. FSAR Section 6.2.

B 3.7 PLANT SYSTEMS

B 3.7.8 Essential Service Water (ESW) System

BASES

BACKGROUND

The ESW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, and a normal shutdown, the ESW System also provides this function for the associated safety related and nonsafety related systems. The safety related function is covered by this LCO.

The ESW System consists of four separate safety related, cooling water trains. Each train consists of one mechanical draft cooling tower, associated basin, pump, piping, valving, instrumentation, and mechanical filtration. Each safety related 2-cell seismic Category I mechanical draft cooling tower rejects energy from the ESW fluid to the ambient and returns the cooled fluid to the ESW cooling tower basin, from which the ESW pumps take suction. Each ESW cooling tower basin is sized for 3 days of post loss of coolant accident (LOCA) operation and ensures adequate volume for the required net positive suction head (NPSH) for the associated ESW pump. Post LOCA evaporative losses are replenished by a safety related seismic Category I source of makeup water. The train associated safety related make-up source delivers water to each basin at ≥ 300 gpm to maintain the NPSH for the ESW pump for up to 30 days following a LOCA. The system pumps and valves are remote and manually aligned, except in the unlikely event of a LOCA or loss of offsite power. The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions.

The mechanical draft cooling towers and basins are safety related, seismic Category I structures sized to provide heat dissipation for safe shutdown following an accident. The cooling tower is protected from tornado missiles.

~~[The seismic Category 1 makeup necessary to support 30 days of post accident mitigation is site specific and details are to be provided by the Combined License applicant]~~The seismic Category 1 emergency makeup water supply, to the ESW cooling tower basins, necessary to support 30 days of post accident mitigation is provided by the safety-related Ultimate Heat Sink (UHS) Makeup Water System that draws water from Lake Ontario. Lake Ontario water enters the Intake Structure forebay through two intake tunnels. The UHS Makeup Water System portion of the Intake Structure houses four independent UHS Makeup Water System trains, one for each ESW division. Each train has one pump, a discharge check valve, and a pump

BASES

BACKGROUND (continued)

discharge isolation motor operated valve, all housed in the UHS Makeup Water Intake Structure, plus the buried piping running up to and into the ESW pumphouse at the ESW cooling tower basin. Each UHS Makeup Water System pump is rated at 750 gpm.

Additional information about the design and operation of the ESW System along with a list of the components served, is presented in FSAR Section 9.2.1 (Ref. 1). The principal safety related functions of the ESW System is the removal of decay heat from the reactor and reactor coolant pump thermal barrier cooling via the Component Cooling Water (CCW) System and removal of operational heat from the emergency diesel generator (EDG).

APPLICABLE SAFETY ANALYSES

The design basis of the ESW System is for two ESW trains, in conjunction with the CCW System, to remove core decay heat and support containment cooling following a design basis LOCA as discussed in FSAR Section 6.2 (Ref. 2). This maintains the In-containment Water Storage Tank fluid within acceptable limits following a LOCA as it is supplied to the Reactor Coolant System by the Emergency Core Cooling System pumps. The ESW System also provides cooling to the train EDG during an anticipated operational occurrence (AOO) or postulated accident.

The ESW System, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR), as discussed in FSAR Section 5.4.7 (Ref. 3), entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CCW and RHR loops that are operating. Two ESW trains are sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ESW System temperature of 95°F occurring simultaneously with maximum heat loads on the system.

Each ESW basin is sized for 3 days of post LOCA operation without requiring makeup. ESW basin makeup is required to maintain NPSH for the ESW pumps beyond 3 days. This volume of water is assumed to be at ≤ 90°F during normal plant operation to prevent exceeding the maximum ESW temperature during a LOCA.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure. The ESW cooling tower and basin is designed in accordance with Regulatory Guide 1.27 (Ref. 4), which requires a 30 day supply of cooling water in the ESW basin, or equivalent make-up.

The ESW System satisfies Criterion 2 and 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The ESW System consists of four trains. Four ESW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

An ESW train is considered OPERABLE when two cooling tower fans, pump, associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE and the ESW basin contains ≥ 27.2 feet of water at ≤ 90°F with capability from makeup from ~~Operable~~ OPERABLE source. ~~[COL applicant to provide definition of OPERABLE makeup source.]~~An OPERABLE emergency makeup water source consists of one OPERABLE train of the UHS Makeup Water System capable of providing makeup water to its associated ESW cooling tower basin. Each UHS Makeup Water System train includes a pump, valves, piping, instruments and controls to ensure the transfer of the required supply of water from Lake Ontario to its associated ESW cooling tower.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ESW System and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ESW System are determined by the systems it supports.

BASES

ACTIONS

The actions have two Notes added. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable ESW train results in an inoperable EDG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ESW train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

A.1

If one ESW train is inoperable, action must be taken to restore OPERABLE status within 120 days. In this condition, the remaining OPERABLE ESW trains are adequate to perform the heat removal function.

The 120 day Completion Time to restore an ESW train to OPERABLE is reasonable since its operation is not assumed in the safety analysis to mitigate the consequences of postulated accidents or AOOs, it provides a reasonable time for repairs, and the low probability of a postulated accident or AOO occurring during this period.

B.1

If two ESW trains are inoperable, action must be taken to restore one to OPERABLE status within 72 hours. In this condition, the two remaining OPERABLE ESW train are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in one of the OPERABLE ESW trains could result in loss of ESW System function.

The 72 hour Completion Time is based on the redundant capabilities afforded by the two OPERABLE trains, and the low probability of a postulated accident occurring during this time period.

C.1 and C.2

If an ESW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that adequate short term (3 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the ESW pumps during the first 3 days post LOCA. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the ESW basin water level is ≥ 27.2 feet from the bottom of the basin.

SR 3.7.8.2

This SR verifies that the ESW System is available to cool the CCW System and EDG to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a postulated accident. With water temperature of the ESW basin ≤ ~~[~~90~~]~~°F, the design basis assumption associated with initial ESW temperature are bounded. With the water temperature of the ESW basin > 90°F, long term cooling capability of the Emergency Core Cooling System (ECCS) loads and Diesel Generators (DGs) may be affected. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.8.3

This SR is modified by a Note indicating that the isolation of the ESW components or systems may render those components inoperable, but does not affect the OPERABILITY of the ESW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the ESW flow path provides assurance that the proper flow paths exist for ESW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.4

Operating each cooling tower fan for ≥ 15 minutes in all speed settings verifies that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the ESW cooling tower fans occurring between surveillances.

SR 3.7.8.5

This SR verifies proper automatic operation of the ESW valves on an actual or simulated actuation signal. The ESW System is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.6

This SR verifies proper automatic operation of the ESW pumps and cooling tower fans on an actual or simulated actuation signal. The ESW System is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.7

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified makeup flowrate ensures that sufficient NPSH can be maintained to operate the ESW pumps following the first 3 days post LOCA. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. This SR verifies that the ESW makeup flowrate is ≥ 300 gpm.

REFERENCES

1. FSAR Section 9.2.1.

2. FSAR Section 6.2.

3. FSAR Section 5.4.7.

4. Regulatory Guide 1.27.

B 3.7 PLANT SYSTEMS

B 3.7.9 Safety Chilled Water (SCW) System

BASES

BACKGROUND

The SCW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, and a normal shutdown, the SCW System also provides this function for the associated safety related systems. The safety related function is covered by this LCO.

The SCW System consists of four independent trains. Each train consists of a chiller refrigeration unit (three 50% compressors per unit), chilled water pumps (two 100% pumps), surge tank, piping, valving, and instrumentation. Heat is rejected to the system chilled water as it passes through the cooling coils of the system users. This heat is rejected from the system as it is pumped through the train chiller refrigeration units. Trains 1 and 4 reject this energy to ambient via air cooled condensers while trains 2 and 3 have condensers cooled by the Component Cooling Water (CCW) System.

The SCW System is normally operating and cools the Control Room Air Conditioning System (CRACS), Safeguards Building Ventilation System Electrical Division (SBVSED), and the train 1 and 4 Low Head Safety Injection (LHSI) pump motor and seal coolers. The combined HVAC function of the SBVSED and SCW systems is backed by a non-safety related, 100% capacity maintenance train which is cooled by the Operational Chilled Water System.

Following a loss of offsite power, previously running SCW trains return to operation once the emergency diesel generator is started and the associated AC electrical power division is re-energized.

The SCW System operation is discussed in FSAR Section 9.2.8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the SCW System is to provide chilled water as a heat sink for the CRAC and SBVSED safety-related HVAC Systems in addition to the LHSI pump motor and seal coolers (train 1 and 4 only). This supports maintaining an acceptable environment in the main control room (MCR) and for safety-related equipment in the essential rooms housing electrical, Instrumentation and Control System, Emergency Feedwater System, and CCW System equipment in the Safeguard Buildings as well as supporting the long term operation of the cooled LHSI pumps in the event of an AOO or postulated accident. Cooling of the electrical rooms requires the availability of each train of SCW in order to ensure the ability of the plant to meet all required safety related functions during any AOO or postulated accident.

BASES

APPLICABLE SAFETY ANALYSES (continued)

A single active failure of a component of the SCW System, with a loss of offsite power, does not impair the ability of the system to perform its design function. The SCW System is designed in accordance with Seismic Category I requirements.

The SCW System satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The SCW System consists of four trains. Four SCW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

An SCW train is considered OPERABLE when one pump, surge tank, the chiller refrigeration unit with two compressors, associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE and in operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SCW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SCW System and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the SCW System are determined by the systems it supports.

ACTIONS

A.1

If one SCW train is inoperable, action must be taken to restore to OPERABLE status within 72 hours. In this condition, the three remaining OPERABLE SCW trains are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in one of the OPERABLE ESW train could result in loss of SCW System function.

The 72 hour Completion Time is based on the redundant capabilities afforded by the three OPERABLE trains, and the low probability of a postulated accident occurring during this time period.

BASES

ACTIONS (continued)

B.1 and B.2

If the SCW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR requires verification every 24 hours that each SCW train is in operation. Verification includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 24 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor SCW train performance.

SR 3.7.9.2

This SR is modified by a Note indicating that the isolation of the SCW components or systems may render those components inoperable, but does not affect the OPERABILITY of the SCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the SCW flow path provides assurance that the proper flow paths exist for SCW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.3

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the in the control room heat load calculation. This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the SCW system is slow and is not expected over this time period.

SR 3.7.9.4

This SR verifies proper automatic operation of the SCW train on an actual or simulated actuation signal. The SCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 9.2.8.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Filtration (CREF)

BASES

BACKGROUND

The CREF provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity~~,~~ ~~[[hazardous chemicals]],~~ or smoke.

The CREF consists of two 100% capacity iodine filtration trains which operate when radioactive contamination is detected at the site or inside the control room envelope (CRE) area. The iodine filtration train is a bypass path of the fresh air intake train for the Control Room Air Conditioning System (CRACS) normal air supply. The air from CRE can also be recirculated through the CREF Iodine Filtration trains. The iodine filtration trains are provided as bypass lines on two of the four normal CRACS air intake trains; other two CRACS intake trains do not have the bypass iodine filtration trains. During an emergency, the fresh outside air and recirculated air are directed through air intake motorized damper and electric heater through the CREF Iodine Filtration train. Each iodine filtration train consists of motorized damper, electric heater, prefilter, upstream HEPA filter, an activated carbon iodine filter, downstream HEPA filter, booster fan, and manual isolation damper. The filtered and clean air is then directed through one or both CRACS normal 75% capacity air conditioning train. Each air conditioning train consists of volume control manual damper, cooling coil, moisture separator, fan suction and discharge silencers, supply air fan, HEPA filter, steam humidifier, non-return damper, volume control electric damper, and fire dampers. The conditioned and clean air is then supplied to the CRE areas. Electric heaters are installed in the CRE supply air ducts to maintain individual room temperatures and relative humidity. The exhaust air from the CRE areas is directed through the recirculation air shaft and then recycled either through the iodine filtration trains or CRACS air conditioning trains. The exhaust from kitchen and sanitary areas is separated from the recycle return air and processed separately.

The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and carbon adsorbers. The HEPA filter bank downstream of the carbon iodine filter collects carbon fines and provides backup in case of failure of the upstream HEPA filter bank. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and carbon adsorbers.

BASES

BACKGROUND (continued)

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CREF train is an emergency system, which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), the outside fresh air supply to the CRE is isolated, and the outside air is directed through the CREF train. The CRE ventilation air is recycled through the air conditioning filter trains and/or CREF train.

Actuation of the CREF places the system in ~~either of two separate states (the~~ emergency radiation ~~state or toxic gas isolation state) of the emergency~~ mode of operation~~, depending on the initiation signal~~. Actuation of the system to the emergency radiation ~~state of the emergency~~ mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the CREF trains. The emergency radiation state also maintains control room pressurization and filtered ventilation of the air supply to the CRE.

Outside makeup air is supplied through the iodine filtration train and added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. ~~The actions taken in the toxic gas isolation state are the same, except that the signal switches the CREF to an isolation alignment to minimize any outside air from entering the CRE through the CRE boundary.~~

The outside air entering the CRE is continuously monitored by radiation ~~[[and toxic gas]]~~ detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic gas isolation state, as required. ~~The actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.~~

BASES

BACKGROUND (continued)

One CREF operating at a flow rate of < 4000 cfm will pressurize the CRE to ≥ 0.125 inches water gauge relative to all external areas adjacent to the CRE boundary. The CREF operation in maintaining the CRE habitability is discussed in FSAR Section 9.4.1 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across one of the other filter trains. Normally open isolation dampers are arranged in series so the failure of one damper to shut will not result in a breach of isolation. The CREF train components are designed in accordance with Seismic Category I requirements.

The CREF is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a postulated accident without exceeding a 5 rem whole body dose or its equivalent to any part of the body 5 rem total effective does equivalent (TEDE).

APPLICABLE SAFETY ANALYSES

The CREF components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREF provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in Chapter 15 (Ref. 2).

The CREF consists of two 100% capacity iodine filtration trains. Each iodine filtration train can be aligned with one of the two 75% capacity air conditioning trains. There are only two iodine filtration trains since only slow failure modes are assumed and filtration efficiency is checked periodically. Both CREF trains with the associated air conditioning trains are required to be OPERABLE. One CREF train is assumed to be lost to a single failure. The other train provides 100% of the ventilation to the CRE.

The CREF provides protection from smoke ~~[[and hazardous chemicals~~ to the CRE occupants. Reference 3 discusses that the need for protection of CRE occupants following a hazardous chemical release~~]]~~ is not required at NMP3NPP. Reference 4 discusses protection of the CRE occupants and their ability to control the reactor from the control room or from the remote shutdown panels in the event of a smoke challenge.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The worst case single active failure of a component of the CREF, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CREF satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

In the event of a postulated accident, one iodine filtration train is required to provide an adequate supply of filtered air to the CRE. To ensure that this requirement is met, both CREF trains must be OPERABLE. The basis for this approach is that two trains are required to satisfy all design requirements (i.e., one train is needed to mitigate the event and other train is assumed to have a single active failure). The failure of both iodine filtration trains could result in exceeding a dose of 5 rem whole body or its equivalent to any part of the body 5 rem TEDE in the event of a large radioactive release.

Each CREF train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CREF train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Prefilters, HEPA filters, and carbon adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CREF trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for postulated accident~~s~~, and that CRE occupants are protected from ~~[[hazardous chemicals and]]~~ smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design conditions, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be procedurelized, and consist of

.

BASES

LCO (continued)

stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CREF trains must be OPERABLE to ensure that the CRE will remain habitable during and following a postulated accident (i.e., LOCA, main steam line break, rod ejection, and fuel handling accident).

In MODE 5 or 6, the CREF is also required to cope with a failure of the Gaseous Waste Processing System.

ACTIONS

A.1

With one CREF train inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the OPERABLE CREF train is adequate to perform the CRE occupant protection function. However, the overall system reliability is reduced. The 7 day Completion Time is based on the low probability of a postulated accident occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1, B.2, and B.3

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of postulated accident consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body 5 rem TEDE), or inadequate protection of CRE occupants from ~~[[hazardous chemicals or]]~~ smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological ~~[[or chemical]]~~ event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a postulated accident, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of postulated accident

BASES

ACTIONS (continued)

consequences, and that CRE occupants are protected from ~~hazardous chemicals and~~ smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a postulated accident occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a postulated accident. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, if any Required Action and Completion Time of Condition A or B cannot be met, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner.

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREF train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREF train in the emergency mode. This action ensures that the other train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

~~Required Action D.1 is modified by a Note indicating to place the system in the toxic gas isolation state with outside air isolated.~~

BASES

ACTIONS (continued)

E.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREF trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the CRE. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

F.1

With both Iodine Filtration trains and associated Air Conditioning trains inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary (i.e., Condition B), the CREF may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations which dry out any moisture accumulated in the carbon from humidity in the ambient air should be performed. Each Iodine filtration train must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy.

SR 3.7.10.2

This SR verifies that the required CREF train testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, carbon adsorber efficiency, minimum flow rate, and the physical properties of the activated carbon. Specific test Frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

This SR verifies that each CREF train starts and operates on an actual or simulated actuation signal. The Frequency of 24 months is based on industry operating experience and is consistent with the typical refueling cycle.

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of postulated accident consequences is no more than 5 rem whole body or its equivalent to any part of the body 5 rem TEDE and the CRE occupants are protected from ~~[[hazardous chemicals and]]~~ smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of postulated accident consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Mitigating actions, or compensatory measures, are discussed in Regulatory Guide 1.196, Section 2.7.3, (Ref. 5) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating measures as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures (Ref. 7). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis postulated accident consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

BASES

REFERENCES

1. FSAR Section 9.4.
2. Chapter 15.
3. FSAR Section 6.4.
4. FSAR Section 9.5.
5. Regulatory Guide 1.196.
6. NEI 99-03, "Control Room Habitability Assessment," March 2003.
7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2005, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability" (ADAMS Accession No. ML040300694).

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Air Conditioning System (CRACS)

BASES

BACKGROUND	The CRACS provides temperature control for the control room envelope (CRE) following isolation of the control room. The CRACS operates in the recycling mode with fresh outside air makeup. There are four normal system 75% capacity identical fresh air intake trains. For each intake train, the fresh air is taken from outside through motorized damper, electric heater, and prefilter. The fresh filtered air is then mixed with the CRE recycled air. The mixed air is then directed through one of the four 75% capacity associated air conditioning train. Each air conditioning train consists of volume control manual damper, cooling coil, moisture separator, fan suction and discharge silencers, supply air fan, HEPA filter, steam humidifier, non-return damper, volume control electric damper, and fire dampers. The conditioned air is supplied to the control room envelope (CRE) areas. Electric heaters are installed in the supply air ducts to maintain individual room temperatures and relative humidity. The exhaust air from the control room envelope (CRE) areas is directed through the recirculation air shaft and then recycled through the air conditioning trains upstream of the cooling coils for each train. The exhaust air from the CRE can also be recycled through the CREF Iodine Filtration trains if contamination is detected in the CRE. The exhaust from kitchen and sanitary areas is separated from the recycle return air and processed separately. Two out of four 75% CRACS Air Conditioning trains operating in the recirculation mode with fresh outside makeup air will provide the required temperature in the Main Control Room (MCR) between 65°F to 75°F, and humidity 40% to 60%. The CRACS operation in maintaining the CRE temperature is discussed in FSAR Section 9.4.1 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the CRACS is to maintain the CRE for 30 days of continuous occupancy. There are four CRACS trains with two trains normally in operation. During emergency operation, one train is assumed to be out for maintenance and a second train is assumed lost to single failure. The two OPERABLE CRACS trains maintain the MCR temperature between 65°F to 75°F. Redundant detectors and controls are provided for

BASES

APPLICABLE SAFETY ANALYSES (continued)

control room temperature control. The CRACS is designed in accordance with Seismic Category I requirements. The CRACS is capable of removing sensible and latent heat loads from the CRE, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

The CRACS satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Four independent and redundant trains of the CRACS are required to be OPERABLE to ensure that at least two are available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CRACS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in all three trains. These components include the heating and cooling coils, moisture separators, humidifiers, and associated temperature control instrumentation. In addition, the CRACS must be operable to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies, the CRACS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

ACTIONS

A.1

With one or two CRACS train(s) inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRACS trains are adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CRACS train could result in loss of CRACS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or non-safety related cooling means are available.

BASES

ACTIONS (continued)

B.1 and B.2

If any Required Action and Associated Completion Time of Condition A is not met in MODE 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

In MODE 5 or 6, or during movement of irradiated fuel, if the inoperable CRACS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRACS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with three or more CRACS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

If three or more CRACS trains are inoperable in MODE 1, 2, 3, or 4, the CRACS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.11.1 This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the in the control room heat load calculation. This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the CRACS is slow and is not expected over this time period.
REFERENCES	1. FSAR Section 9.4.1.

B 3.7 PLANT SYSTEMS

B 3.7.12 Safeguard Building Controlled Area Ventilation System (SBVS)

BASES

BACKGROUND

The SBVS provides a protected environment in the hot mechanical areas of Safeguards Building Divisions 1, 2, 3, and 4 and the fuel building. The SBVS also filters airborne radioactive particulates from the areas of the active Emergency Core Cooling System (ECCS) components during a Loss of Coolant Accident (LOCA).

The conditioned air supply to all four Safeguard Building Divisions is provided independently for each division by the Electrical Division of Safeguard Ventilation System (Ref. 1). The SBVS supplies the conditioned air for ventilation through a volume control damper and two isolation dampers for each division to the hot mechanical areas of the four Safeguard Building Divisions. The SBVS air supply and exhaust flows are designed to prevent spread of airborne contamination and to maintain a negative pressure in the safeguard buildings and fuel building.

Under normal plant operation, the operational air exhaust from each hot area is drawn independently through a volume control damper and two isolation dampers located on the operational exhaust duct system for each safeguard building. The main exhaust duct of each division is connected to a common concrete duct which runs inside the annulus. The operational air exhaust is then drawn through a concrete duct cell for processing by the normal filtration train of the Nuclear Auxiliary Building Ventilation System prior to release through the plant stack (Ref. 2).

During conditions in which a release of airborne contamination from any of the four hot mechanical areas occurs, the SBVS will redirect the accident air exhaust independently via four separate exhaust lines which join into one common leak-tight exhaust duct inside the annulus. The exhaust duct then connects to an accident exhaust filtration train located in the fuel building. There are two 100% capacity accident iodine exhaust filtration trains in parallel configuration. Each train consists of inlet motor controlled damper, electric heater, pre-filter, upstream HEPA filter, iodine filter with activated carbon, downstream HEPA filter, outlet motor controlled damper, exhaust fan, and non-return damper. The accident air exhaust is processed through one or both independent iodine filtration trains prior to release through the plant stack. The downstream HEPA filter is not credited in the analysis, but serves to collect carbon particles and provides a backup in case the upstream HEPA filter bank fails. The pre-filters remove any large particles in the air and any entrained water droplets present to prevent excessive loading of the HEPA filters and carbon adsorbers.

BASES

BACKGROUND (continued)

In case of a LOCA with assumed ECCS leakage, the accident air exhaust from the safeguard buildings and fuel building is also directed through the accident iodine exhaust filtration trains prior to release through the plant stack.

The SBVS accident iodine filtration train is a standby system which may also be operated during normal plant operations. Upon receipt of an actuating signal, the normal air exhaust from the buildings is isolated and the accident air is redirected through the iodine filtration train.

The SBVS is discussed in FSAR Section 9.4.5 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The SBVS design basis is established by the consequences of the limiting postulated accident, which is a LOCA with assumed ECCS leakage. The analysis of a LOCA, given in Reference 4, assumes ECCS leakage to the safeguard buildings and fuel building is a conservative four gallons a minute. The SBVS consists of two 100% capacity iodine filtration trains in parallel configuration. There are only two iodine filtration trains since only slow failure modes are assumed and filtration efficiency is checked periodically. Both sets of iodine filtration trains are required to be OPERABLE. One SBVS train is then assumed to be lost due to a single failure. The postulated accident analysis assumes that two trains of the SBVS are OPERABLE. The accident analysis accounts for the reduction in airborne radioactive material provided by the one train of this filtration system. The amount of fission products available for release from the safeguard buildings and fuel building is determined for a LOCA. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 5).

The SBVS is not credited in the Fuel Handling Accident evaluation.

The SBVS satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Two independent and redundant trains of SBVS Accident Exhaust Filtration are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power.

The failure of both trains could result in the atmospheric release from the safeguard buildings and fuel building exceeding the 10 CFR 50.34 (Ref. 6) limits in the event of a LOCA.

BASES

LCO (continued)

The SBVS Accident Exhaust Filtration train is considered OPERABLE when it's associated:

a. Fan is OPERABLE;

b. Prefilter, HEPA filter and carbon adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

c. Heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The LCO is modified by a Note allowing the safeguard building and fuel building boundaries to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for safeguard building or fuel building isolation is indicated.

APPLICABILITY

In MODE 1, 2, 3, or 4, the SBVS Accident Exhaust Filtration train is required to be OPERABLE to provide fission product removal associated with the leakage inside the hot areas of the Safeguard Buildings.

In MODE 5 or 6, the SBVS Accident Exhaust Filtration train is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one SBVS Accident Exhaust Filtration train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the SBVS function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable SBVS train, and the remaining SBVS train providing the required protection.

BASES

ACTIONS (continued)

B.1

~~---REVIEWER'S NOTE--------------------------------~~
~~Adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of an intentional and unintentional entry into Condition B.~~
~~--~~

If the safeguard buildings or fuel building boundary is inoperable in MODE 1, 2, 3, or 4, the SBVS trains may not be able to perform their intended functions. Actions must be taken to restore an OPERABLE safeguard buildings and fuel building boundaries within 24 hours. During the period that the safeguard buildings or fuel building boundary is inoperable, appropriate compensatory measures consistent with the intent, as applicable, of GDC 19 and 10 CFR Part 100 ~~should~~ shall be utilized to protect plant personnel from potential hazards such as radioactive contamination,~~[[toxic chemicals,]]~~ smoke, temperature and relative humidity, and physical security. Preplanned measures ~~should~~ shall be available and implemented upon entry into the condition to address these concerns ~~for~~ regardless of whether the entry is intentional ~~and~~ or unintentional entry ~~into the condition~~. The 24 hour Completion Time is reasonable based on the low probability of a postulated accident occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the safeguard buildings or fuel building boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the associated Completion Time, or when both SBVS Accident Exhaust Filtration trains are inoperable for reasons other than an inoperable safeguard building or fuel building boundary (i.e., Condition B), the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Verifying that safeguards building and fuel building negative pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to safeguards building and fuel building pressure variations and pressure instrument drift during the applicable MODES.

SR 3.7.12.2

Maintaining safeguards building and fuel building OPERABILITY requires verifying each access opening door is closed. However, all safeguards building and fuel building access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.7.12.3

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Monthly heater operation dries out any moisture accumulated in the carbon from humidity in the ambient air. Systems with heaters must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.12.4

This SR verifies that the required SBVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, carbon adsorber efficiency, minimum system flow rate, and the physical properties of the activated carbon (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.5

This SR verifies that each SBVS train starts and operates on an actual or simulated actuation signal. The 24 month Frequency is consistent with Reference 7.

SR 3.7.12.6 and 3.7.12.7

The SBVS exhausts the safeguards building and fuel building atmosphere to the environment through appropriate treatment equipment. Each safety SBVS train is designed to draw down the safeguards building and fuel building to a negative pressure ≥ 0.25 inches of water gauge (wg) in ≤ 305 seconds and maintain the safeguards building and fuel building at a negative pressure ≥ 0.25 inches wg at a flow rate ≤ 2,400 cfm from the safeguards building and fuel building. To ensure that all fission products released to the safeguards building and fuel building are treated, SR 3.7.12.6 and SR 3.7.12.7 verify that a pressure in the safeguards building and fuel building that is less than the lowest postulated pressure external to the safeguards building and fuel building boundaries can be established and maintained. When the SBVS is operating as designed, the establishment and maintenance of safeguards building and fuel building pressure cannot be accomplished if the safeguards building or fuel building boundaries is not intact. Establishment of this pressure is confirmed by SR 3.7.12.6. SR 3.7.12.7 demonstrates that the safeguards building and fuel building can be maintained at a negative pressure ≥ 0.25 inches wg. The primary purpose of these SRs is to ensure safeguards building and fuel building boundary integrity. The secondary purpose of these SRs is to ensure that the SBVS train being tested functions as designed. There is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the SBVS. These SRs need not be performed with each safety SBVS train. The SBVS train used for these Surveillances is staggered to ensure that in addition to the requirements of LCO 3.7.12, either safety SBVS train will perform this test. The inoperability of the SBVS does not necessarily constitute a failure of these Surveillances relative to the safeguards building and fuel building OPERABILITY. Operating experience has shown the safeguards building and fuel building boundaries usually pass these Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

REFERENCES

1. FSAR Section 9.4.6.
2. FSAR Section 9.4.3.
3. FSAR Section 9.4.5.
4. FSAR Section 15.0.
5. Regulatory Guide 1.25.
6. 10 CFR 50.34.
7. Regulatory Guide 1.52, Rev. 3.

B 3.7 PLANT SYSTEMS

B 3.7.13 Safeguards Building Ventilation System Electrical Division (SBVSED)

BASES

BACKGROUND	The SBVSED provides temperature control for the electrical and instrumentation and control rooms of each safeguards building. The SBVSED consists of four independent trains that provide cooling and heating of the electrical equipment areas of each safeguards building. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for temperature control. The SBVSED can be operated with or without recycled air depending on the outside air temperature. The SBVSED is an emergency system which also operates during normal unit operations and accident conditions to provide ventilation and cooling in the electrical equipment areas of the safeguards buildings. Following a loss of offsite power, previously running SBVSED trains return to operation once the emergency diesel generator is started and the associated AC electrical power division is re-energized. The SBVSED operation in maintaining the safeguards building temperature is discussed in FSAR Section 9.4.6 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the SBVSED is to provide ventilation and air conditioning to the electrical equipment area of the safeguards buildings following any Design Basis Accident (DBA). There are four SBVSED trains, with one train normally in operation in each of the four safeguards buildings. During emergency operation, one train is assumed to be lost to single failure of a diesel generator. The three OPERABLE SBVSED trains maintain their respective safeguards building in a pre-determined temperature range. The SBVSED is designed in accordance with Seismic Category I requirements. The SBVSED is capable of removing sensible and latent heat loads from the safeguards building, which include consideration of equipment heat loads, to ensure equipment OPERABILITY. The SBVSED satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).
LCO	Four independent trains of the SBVSED are required to be OPERABLE and in operation to ensure that at least three are available, assuming a single failure disabling one train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

BASES

LCO (continued)

The SBVSED is considered to be OPERABLE when the individual components necessary to maintain the safeguards building temperature are OPERABLE and in operation in all four trains. These components include the cooling coils and associated temperature control instrumentation.

APPLICABILITY

In MODES 1, 2, 3, 4, the SBVSED must be OPERABLE and in operation to ensure that the safeguards building electrical equipment areas will not exceed equipment operational requirements following a DBA.

In MODES 5 and 6, the OPERABILITY requirements of the SBVSED is determined by the systems that it supports.

ACTIONS

A.1

With one SBVSED train inoperable or not in operation, action must be taken to restore OPERABLE status within 72 hours. In this condition, the remaining OPERABLE SBVSED trains are adequate to maintain the three remaining safeguards building temperature within limits. A non-safety maintenance train is available to provide temperature control in the affected safeguards building electrical area. However, the overall reliability is reduced because a loss of offsite power would result in loss of SBVSED function in the affected train. The 72 hour Completion Time is based on the low probability of an event occurring, the consideration that the remaining safeguards trains can provide the required safety function, and that alternate, non-safety related cooling means are available.

B.1 and B.2

If any Required Action and associated Completion Time of Condition A is not met, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

Each SBVSED train is verified to be in operation at a frequency of 24 hours to verify that ventilation and air conditioning to the electrical equipment area of each safeguard building. The 24 hour Frequency is appropriate since the train is normally in operation and other indications are available to alert the control room to a failure of a SBVSED train.

SR 3.7.13.2

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the in the safeguards building heat load calculation. This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the SBVSED is slow and is not expected over this time period.

SR 3.7.13.3

This SR verifies proper automatic operation of the SBVSED train on an actual or simulated actuation signal. The SBVSED System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 9.4.6.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the spent fuel storage pool design is given in FSAR Section 9.1.2 (Ref. 1). A description of the Fuel Pool Cooling and Purification System is given in FSAR Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in FSAR Section 15.7.4 (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the spent fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is well within the limits of Table 6 of Regulatory Guide 1.183 (Ref. 5). According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop. The fuel storage pool water level satisfies Criteria 2 of 10 CFR 50.36(d)(2)(ii).
LCO	The spent fuel storage pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

BASES

APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool, since the potential for a release of fission products exists.
ACTIONS	A.1 Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel storage pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	SR 3.7.14.1 This SR verifies sufficient spent fuel storage pool water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience. During refueling operations, the level in the spent fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.
REFERENCES	1. FSAR Section 9.1.2. 2. FSAR Section 9.1.3. 3. FSAR Section 15.7.4. 4. Regulatory Guide 1.25, March 1972. 5. Regulatory Guide 1.183, Table 6, July 2000.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Storage Pool Boron Concentration

~~-----Reviewer's Note-----~~
~~The design of the spent fuel storage racks is to be provided by the COLA applicant. The required boron concentration will be provided as a part of the spent fuel rack design.~~
~~-----~~

BASES

BACKGROUND

The water in the spent fuel storage pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel in the spent fuel storage racks, the design basis for preventing criticality outside the reactor is that there is a 95 percent probability at a 95 percent confidence level, without soluble boron, that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 0.995, including uncertainties and tolerances. The NRC guidelines specify a limiting k_{eff} of 1.0 for normal storage in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains a subcritical condition for the allowed loading patterns. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single independent accident need be considered at one time. For example, the only accident scenario that has a potential for more than negligible positive reactivity effect is an inadvertent misplacement of a new fuel assembly. This accident has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the location of each assembly in accordance with LCO 3.7.16, "Spent Fuel Pool Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.~~As described in the following LCO 3.7.16, "Spent Fuel Storage," fuel assemblies are stored in the spent fuel racks without restriction. Although the water in the spent fuel pool is normally borated to ≥ [1291] ppm with boric acid enriched to ≥ 37% B^{10}, the criteria that limit storage of a fuel assembly to specific locations are conservatively developed without taking credit for boron.~~

APPLICABLE SAFETY

Although credit for the soluble boron normally present in the spent fuel ~~----~~
~~-----REVIEWER'S NOTE-----~~
pool water is permitted under abnormal or accident conditions, most ~~The design of the spent fuel storage racks is the responsibility of the COL~~
SES abnormal or accident conditions will not result in exceeding the limiting ~~applicant. A COL applicant that references the U.S. EPR design certification will demonstrate that the design satisfies the criticality analysis requirements for the spent fuel storage racks.~~

reactivity even in the absence of soluble boron. The effects on reactivity of credible abnormal and accident conditions due to temperature increase, boiling, assembly dropped on top of a rack, lateral rack module movement and misplacement of a spent fuel assembly have been analyzed. The spent fuel pool k_{eff} storage limit of 0.95 is maintained during these events by a minimum boron concentration of 500 ppm with boric acid enriched to ≥ 37% B^{10} established by criticality analysis (Ref. 2). Compliance with the LCO minimum boron concentration limit of 500 ppm with boric acid enriched to ≥ 37% B^{10} ensures that the credited concentration is always available.

The concentration and enrichment of dissolved boron in the spent fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO

The spent fuel storage pool boron concentration is required to be ≥ ~~[1291]~~500 ppm boron enriched to ≥ 37% B^{10}. The specified concentration and enrichment of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 2~~1~~. This concentration of dissolved boron is the minimum required concentration and enrichment for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1, and A.2.2

When the concentration or enrichment of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration or enrichment of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the spent fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration and enrichment of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

~~ACTIONS~~

~~A.1, A.2.1, and A.2.2~~

~~When the concentration or enrichment of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration or enrichment of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration and enrichment of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.~~

~~The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR and SR 3.7.15~~3~~.2 are met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

SR 3.7.15.2

Verification every 24 months that the B^{10} enrichment is within limit ensures that the B^{10} concentration assumed in the accident analyses is available. Since the boron in the spent fuel pool is not exposed to a significant neutron flux, 24 months is considered conservative.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2).

2. ~~[Holtec Topical Report]~~UN-TR-08-001, “Spent and New Fuel Storage Analyses for U.S. EPR Topical Report,” March 2008.~~ANSI/ANS 8.1-1998 "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."~~

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Storage

~~---Reviewer's Note---~~
~~The design of the spent fuel storage racks is to be provided by the COLA applicant. The required spent fuel storage configuration will be provided as a part of the spent fuel rack design.~~
~~---~~

BASES

BACKGROUND

The high density spent fuel storage racks are divided into two separate and distinct regions as shown in Figure 4.3-1. Region 1, with a maximum of 360 storage locations, is designed to accommodate new fuel assemblies with a maximum enrichment of 5.0 weight percent U-235, or spent fuel assemblies regardless of the combination of initial enrichment and burnup. Region 2, with a maximum of 1000 storage locations, is designed to accommodate spent fuel assemblies in all locations which comply with the combination of initial enrichment and burnup limits specified in Figure 3.7.16-1, Fuel Assembly Burnup Requirements for Region 2.

The water in the spent fuel storage pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel in the spent fuel racks, the design basis for preventing criticality outside the reactor is that there is a 95 percent probability at a 95 percent confidence level, without soluble boron, that the effective multiplication faction (k_{eff}) of the fuel assembly array will be less than 0.995, including uncertainties and tolerances. The NRC guidelines specify a limiting k_{eff} of 1.0 for normal storage in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains a subcritical condition for the allowed loading patterns.

The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal and accident conditions, since only a single independent accident need be considered at one time. For example, the only accident scenario that has the potential for more than negligible positive reactivity effect is an inadvertent misplacement of a new fuel assembly. This accident has is a potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, enriched boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the combination of initial enrichment and burnup of the stored fuel in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.

BASES~~The spent fuel storage facility as described in Ref. 1 has a capacity of at least [1020] fuel assemblies. The spent fuel storage racks are designed to accommodate fuel with a maximum nominal enrichment of 5.0 wt% U-235 (maximum tolerance of +/- 0.05 wt%). The spent fuel storage racks are designed for unrestricted spent fuel assembly storage.~~

APPLICABLE SAFETY ANALYSES

The hypothetical accidents can only take place during or as a result of the movement of an assembly (Refs. 2 and 3). For these accident ~~The~~ occurrences, the presence of soluble boron in the spent fuel storage pool

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~

~~design of the spent fuel storage racks is the responsibility of the COL applicant. A COL applicant that references the U.S. EPR design certification will demonstrate that the design satisfies the criticality analysis requirements for the spent fuel storage racks.~~

~~---~~

(controlled by LCO 3.7.15, “Spent Fuel Pool Boron Concentration”) prevents criticality. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.~~The criticality analysis shows that the fuel remains subcritical under all credible abnormal conditions.~~

~~The design shows acceptable prevention of an increase in effective multiplication factor (k-effective) beyond safe limits based on the guidelines in Ref. 2.~~

The configuration of fuel assemblies in the spent fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within Region 2 of the spent fuel pool in the accompanying LCO, ensure the k_{eff} of the spent fuel storage pool will always remain < 0.995, assuming the pool to be flooded with unborated water and < 0.95, with a boron concentration of greater than 500 ppm and boron enrichment ≥ 37%.

Storage of spent fuel is permitted in all Region 2 locations provided that the spent fuel meets the combination of initial enrichment and burnup requirements shown in Figure 3.7.16-1, Fuel Assembly Burnup Requirements for Region 2.~~Unrestricted storage of fuel assemblies within the spent fuel pool is allowed provided that the maximum nominal U-235 enrichment is equal to or less than 5.00 weight percent. This ensures the k-effective of the spent fuel pool will always remain less than 0.95 assuming the pool is flooded with [unborated water].~~

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 of the spent fuel storage pool.

ACTIONS

A.1

When the requirements of the LCO are not met, action must be immediately initiated to move the non-complying fuel assembly to an acceptable storage location (i.e., Region 1).

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

~~ACTIONS~~ ~~A.1~~

~~When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with the LCO, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance.~~

~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that ~~the fuel assembly is in accordance with the configurations specified in the accompanying LCO~~ initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1 prior to storing the fuel assembly in Region 2.

REFERENCES

1. Double contingency principle ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2).~~FSAR Section 9.1.2~~

2. ~~2.~~ UN-TR-08-001, “Spent and New Fuel Storage Analyses for U.S. EPR Topical Report,” March 2008.~~[Holtec Topical Report]10CFR 50.68, "Criticality Accident Requirements."~~

3. U.S. EPR FSAR Section 15.0.3.10, “Fuel Handling Accident.”

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND	Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant. A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents. This limit bounds the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.12, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.15, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours). With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power. Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 50.34 (Ref. 1) limits.
APPLICABLE SAFETY ANALYSES	The accident analysis of the main steam line break (MSLB), as discussed in FSAR Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates.

BASES

APPLICABLE SAFETY ANALYSES (continued)

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and Main Steam Relief Trains (MSRTs). The Emergency Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Low Head Safety Injection System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and MSRTs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a postulated accident to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a postulated accident.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS	A.1 and A.2 DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.17.1 This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.
REFERENCES	1. 10 CFR 50.34. 2. FSAR Chapter 15.

Section B – {NMP3NPP} Technical Specifications and Bases

A complete copy of the {NMP3NPP} Technical Specifications and Bases are provided in {the following pages in} this section. This copy incorporates the plant specific information and values, removes Reviewer’s Notes and incorporates the departures and supplements addressed in Section A of this part of the COL Application.

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1.0 USE AND APPLICATION

1.1 Definitions

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATING DEVICE OPERATIONAL TEST (ADOT)	A ADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the division required for trip actuating device OPERABILITY. The ADOT may be performed by means of any series of sequential, overlapping, or total division steps.
AXIAL OFFSET (AO)	AO shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core. $AO = ((Upper - Lower) \ / \ (Lower + Upper)) * 100$
AZIMUTHAL POWER IMBALANCE (API)	AZIMUTHAL POWER IMBALANCE shall be the maximum of the difference between the maximum power generated in any core quadrant (QN_{max}) and the minimum power generated in any core quadrant (QN_{min}), as measured by the power range excore detectors. $API = QN_{max} - QN_{min}$
CALIBRATION	A CALIBRATION shall be the adjustment, as necessary, of the sensor output such that it responds within the necessary range and accuracy to known values of the parameter that the sensor monitors. The CALIBRATION shall encompass all devices in the division required for sensor OPERABILITY. CALIBRATION of instrument divisions with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal CALIBRATION of the remaining adjustable devices in the division. The CALIBRATION may be performed by means of any series of sequential, overlapping, or total steps.

1.1 Definitions

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status to other indications or status derived from independent instrumentation channels measuring the same parameter.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.
DIVISION OPERATIONAL TEST (DOT)	A DOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for OPERABILITY. The DOT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for division OPERABILITY such that the setpoints are within the necessary range and accuracy. The DOT may be performed by means of any series of sequential, overlapping, or total steps.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from EPA Federal Guidance Report No. 11, 1988, "imiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

1.1 Definitions

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil" or the average gamma disintegration energies as provided in ICRP Publication 38, "Radionuclide Transformations" or similar source.
EXTENDED SELF TESTS	Testing of the Protection System signal processors that cannot be performed during power operation are performed during the start-up of a computer. These tests can also be initiated by pushing a reset button on the computer. These tests include a basic hardware test using the internal diagnostics monitor, a self-test of the operating system, and basic hardware tests.

1.1 Definitions

LEAKAGE	LEAKAGE shall be: a. Identified LEAKAGE 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System; b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; and c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

1.1 Definitions

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in FSAR Chapter 14, "Verification Programs"; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the low temperature overpressure protection setpoints, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)."
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 4590 MWt.
SENSOR OPERATIONAL TEST (SOT)	A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the input circuit required for OPERABILITY. The SOT shall include the verification of the accuracy and time constants of the analog input modules. The SOT may be performed by means of any series of sequential, overlapping, or total steps.

1.1 Definitions

SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, trains, divisions, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, trains, divisions, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, divisions, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	$\geq$ 0.99	> 5	NA
2	Startup	$\geq$ 0.99	$\leq$ 5	NA
3	Hot Standby	< 0.99	NA	$\geq$ 350
4	Hot Shutdown [b]	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown [b]	< 0.99	NA	$\leq$ 200
6	Refueling [c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

<u>EXAMPLE 1.2-1</u>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . <u>AND</u> A.2 Restore . . .	

In this example the logical connector <u>AND</u> is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

However, when a <u>subsequent</u> train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the <u>first</u> inoperability; and

1.3 Completion Times

DESCRIPTION (continued)

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ."

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

1.3 Completion Times

EXAMPLES (continued)

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

1.3 Completion Times

EXAMPLES (continued)

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X division inoperable.	A.1 Restore Function X division to OPERABLE status.	7 days
B. One Function Y division inoperable.	B.1 Restore Function Y division to OPERABLE status.	72 hours
C. One Function X division inoperable. AND One Function Y division inoperable.	C.1 Restore Function X division to OPERABLE status. OR C.2 Restore Function Y division to OPERABLE status.	72 hours 72 hours

When one Function X division and one Function Y division are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each division starting from the time each division was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second division was declared inoperable (i.e., the time the situation described in Condition C was discovered).

1.3 Completion Times

EXAMPLES (continued)

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A.

It is possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4	6 hours 12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

1.3 Completion Times

EXAMPLES (continued)

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

EXAMPLE 1.3-5

ACTIONS

--NOTE--
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

1.3 Completion Times

EXAMPLES (continued)

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

<u>EXAMPLE 1.3-6</u>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One division inoperable.	A.1 Perform SR 3.x.x.x. <u>OR</u> A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

1.3 Completion Times

EXAMPLES (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

<u>EXAMPLE 1.3-7</u>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated. <u>AND</u> A.2 Restore subsystem to OPERABLE status.	1 hour <u>AND</u> Once per 8 hours thereafter 72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

1.3 Completion Times

EXAMPLES (continued)

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements. Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction. The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria. Some Surveillances contain Notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

1.4 Frequency

DESCRIPTION (continued)

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered;

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CALIBRATION	24 months

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (24 months) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 24 months, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

1.4 Frequency

EXAMPLES (continued)

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE-------------------------- Not required to be performed until 12 hours after ≥ 25% RTP. --- Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance was not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance was not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE-------------------------- Only required to be met in MODE 1. --- Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency was not met), SR 3.0.4 would require satisfying the SR.

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE-------------------------- Only required to be performed in MODE 1. --- Perform complete cycle of the valve.	7 days

1.4 Frequency

EXAMPLES (continued)

The interval continues, whether or not the unit operation is in MODES 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance was not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance was not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE-------------------------- Not required to be met in MODE 3. --- Verify parameter is within limits.	 24 hours

1.4 Frequency

EXAMPLES (continued)

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency was not met), SR 3.0.4 would require satisfying the SR.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.246 when using the ACH-2 DNB correlation and ≥ 1.21 when using the BWU-N correlation; and

2.1.1.2 The peak fuel centerline temperature shall be < 4901°F, decreasing by 14°F per 10,000 MWD/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be ≤ 2803 psia.

2.2 SL VIOLATIONS

2.2.1 If SL 2.1.1.1 or 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1

LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8.

LCO 3.0.2

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

LCO 3.0.3

When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or

3.0 LCO Applicability

LCO 3.0.4 (continued)

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7 Test Exception LCO 3.1.9, "PHYSICS TESTS Exceptions – MODE 2," and LCO 3.4.17, "RCS Loops – Test Exceptions," allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met.

When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0 LCO Applicability

LCO 3.0.8

When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. The snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. The snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

3.0 SR Applicability

SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM to be within the limits specified in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within ± 1000 pcm of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	7 days
	AND	
	A.2 Establish appropriate operating restrictions and SRs.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	--------------------------NOTE--------------------------------- The predicted reactivity values must be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. --- Verify measured core reactivity is within ± 1000 pcm of predicted values.	Once prior to entering MODE 1 after each refueling <u>AND</u> --------NOTE-------- Only required after 60 EFPD ------------------------ 31 EFPD thereafter

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR and a maximum upper limit as specified below:

a. 5 pcm/°F when THERMAL POWER < 50% RTP; and

b. 0 pcm/°F when THERMAL POWER is ≥ 50% RTP.

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit, MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. MTC not within upper limit.	A.1	Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 2 with keff < 1.0.	6 hours
C. MTC not within lower limit.	C.1	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MTC is within upper limit.	Once prior to entering MODE 1 after each refueling
SR 3.1.3.2	-------------------------------NOTE------------------------------ If the MTC is more negative than the COLR limit when extrapolated to the end of cycle, SR 3.1.3.2 must be repeated prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. --- Verify the MTC is within the lower limit specified in the COLR.	Once each fuel cycle within 7 EFPD of reaching 2/3 of expected core burnup

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Control Cluster Assembly (RCCA) Group Alignment Limits

LCO 3.1.4 All shutdown and control RCCAs shall be OPERABLE.

AND

Individual indicated analog RCCA positions shall be within 8 steps of their group digital RCCA position indication.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCCAs inoperable.	A.1.1 Verify SDM to be within the limits specified in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours
B. One RCCA not within alignment limits.	B.1.1 Verify SDM to be within the limits specified in the COLR.	1 hour
	OR	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.2 Reduce THERMAL POWER to ≤ 50% RTP. <u>AND</u> B.3 Verify SDM is within the limits specified in the COLR. <u>AND</u> B.4 Perform a flux map, using the Aeroball Measurement System, and calibrate the self powered neutron detectors. <u>AND</u> B.5 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	2 hours Once per 12 hours 12 hours 5 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two or more RCCAs not within alignment limits.	D.1.1 Verify SDM to be within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	D.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual RCCA positions within alignment limit.	12 hours
SR 3.1.4.2	Verify RCCA freedom of movement (trippability) by moving each RCCA not fully inserted in the core $\geq$ 16 steps in either direction.	92 days
SR 3.1.4.3	Verify drop time of each RCCA, from the fully withdrawn position, is $\leq$ 3.5 seconds from opening of the reactor trip breaker to the centerline of lowest RCCA position indication coil, with: a. Tavg $\geq$ 500°F; and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.5 Each shutdown bank shall be within insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

---NOTE---
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1 Verify SDM is within the limits specified in the COLR. OR A.1.2 Initiate boration to restore SDM to within limit. AND A.2 Restore shutdown bank(s) to within limits.	1 hour 1 hour 2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown bank is within the insertion limits specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Each rod control cluster assembly (RCCA) control bank shall be within insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

--NOTE--
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCCA control banks with insertion limits not met.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore RCCA control bank(s) to within insertion limits.	2 hours
B. One or more RCCA control banks with sequence or overlap limits not met.	B.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.2	Restore RCCA control bank(s) to within sequence and overlap limits.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify estimated critical RCCA control bank position is within the limits specified in the COLR.	Once within 4 hours prior to achieving criticality
SR 3.1.6.2	Verify each RCCA control bank position is within the insertion, sequence, and overlap limits specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Control Cluster Assembly (RCCA) Position Indication.

LCO 3.1.7 The Analog and Digital RCCA Position Indication shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each analog RCCA position indicator and each digital RCCA position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more banks with one individual analog RCCA position indicator inoperable.	A.1.1 Implement the imbalance/dropped RCCA penalty on the Low DNBR reactor trip setpoint.	8 hours
	<u>AND</u>	
	A.1.2 Verify position of RCCAs with inoperable position detectors indirectly using the SPNDs.	Once per 8 hours <u>AND</u> Once within 4 hours after an RCCA with an inoperable analog RCCA position indicator has been moved in excess of 20 steps in one direction since the last determination of the RCCA's position
	<u>OR</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. One or more banks with two or more analog RCCA position indicators inoperable.	B.1	Verify three rod control cluster assembly units (RCCAUs) are OPERABLE.	Immediately
	AND		
	B.2	Place the RCCAs under manual control.	Immediately
	AND		
	B.3	Determine the Reactor Coolant System T_{avg}.	Once per 1 hour
	AND		
	B.4	Restore inoperable analog RCCA position indicator(s) to OPERABLE status such that a maximum of one analog RCCA position indicator in the associated bank is inoperable.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more banks with one digital RCCA position indicator inoperable.	C.1.1 Verify by administrative means all analog RCCA position indicators for the affected bank are OPERABLE.	Once per 8 hours
	AND	
	C.1.2 Verify the most withdrawn RCCA and the least withdrawn RCCA of the affected bank are within the RCCA alignment limits.	Once per 8 hours
	OR	
	C.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. Required Action and associated Completion Time of Condition not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify each analog RCCA position indication agrees within 8 steps of the digital RCCA position indication over a span from 10 steps to the full out position that is defined in COLR.	Once prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Boron Dilution Protection (BDP)

LCO 3.1.8 The volume control tank (VCT) and letdown isolation valves shall be OPERABLE.

APPLICABILITY: MODES 3, 4, 5, and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more VCT or letdown isolation valves inoperable.	A.1 Isolate the affected boron dilution flow path by use of at least one closed and de-activated automatic valve or closed manual valve.	8 hours
	<u>AND</u>	
	A.2 -------------NOTE--------------- Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. -------------------------------------	
	Verify the affected boron dilution flow path is isolated.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify the isolation time of each VCT and letdown isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.1.8.2	Verify each VCT and letdown isolation valve actuates to the isolation position on an actual or simulated signal.	24 months

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.9 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient,"
LCO 3.1.4, "RCCA Group Alignment Limits,"
LCO 3.1.5, "Shutdown Bank Insertion Limits," and
LCO 3.1.6, "Control Bank Insertion Limits,"

may be suspended provided:

a. SDM is within the limits specified in the COLR and;

b. THERMAL POWER is $\leq$ 5% RTP.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes
	<u>AND</u>	
	A.2 Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes
SR 3.1.9.2	Verify SDM is within the limits specified in the COLR.	24 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Linear Power Density (LPD)

LCO 3.2.1 The LPD shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LPD not within limits.	A.1 Restore LPD to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 10% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify LPD is within limits specified in the COLR.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 90% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F^N_{\Delta H}$ not within limits.	A.1 Reduce THERMAL POWER by 1% for each 1% that $F^N_{\Delta H}$ exceeds the limits. <u>AND</u> A.2 Restore $F^N_{\Delta H}$ to within limits.	1 hour 4 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER ≤ 90% RTP.	2 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after exceeding 90% power. -- Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling outage prior to exceeding 98% RTP <u>AND</u> 15 effective full power days thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 Departure From Nucleate Boiling Ratio (DNBR)

LCO 3.2.3 The DNBR shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DNBR not within limits.	A.1 Restore DNBR to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 10% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify DNBR is within limits specified in the COLR.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.4 AXIAL OFFSET (AO)

LCO 3.2.4 The AO shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER $\geq$ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AO not within limits.	A.1 Restore AO to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.4.1 Verify AO is within the limits specified in the COLR.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.5 AZIMUTHAL POWER IMBALANCE (API)

LCO 3.2.5 The API shall be maintained ≤ 1.04.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. API not within limit.	A.1 Restore API to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.5.1 Verify API is ≤ 1.04.	12 hours

3.3 INSTRUMENTATION

3.3.1 Protection System (PS)

LCO 3.3.1 The PS sensors, manual actuation switches, signal processors, and actuation devices specified in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

-- NOTE --
Separate Condition entry is allowed for each sensor, manual actuation switch, signal processor, and actuation device.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more sensors inoperable.	A.1 --------------NOTE--------------- Only applicable for Table 3.3.1-1, Component A.21. ------------------------------------- Place inoperable sensor in trip.	1 hour
	<u>AND</u>	
	A.2 --------------NOTE--------------- Not applicable for Table 3.3.1-1, Component A.21. ------------------------------------- Place inoperable sensor in lockout.	4 hours
B. One or more manual actuation switches inoperable.	B.1 Restore manual actuation switch to OPERABLE status.	48 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more acquisition and processing units (APUs) inoperable due to the Setpoint Control Program requirements for one or more Trip/Actuation Functions not met.	C.1 --------------NOTE--------------- Only applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
	Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown," for emergency diesel generator (EDG) made inoperable by inoperable APU.	1 hour
	<u>AND</u>	
	C.2 --------------NOTE--------------- Not applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
	Place the Trip/Actuation Function in the associated APU in lockout.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more signal processors inoperable for reasons other than Condition C.	D.1 --------------NOTE--------------- Only applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
	Enter applicable Conditions and Required Actions of LCO 3.8.1 and LCO 3.8.2 for EDG made inoperable by inoperable APU.	1 hour
	AND	
	D.2 --------------NOTE--------------- Not applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
	Place inoperable signal processor in lockout.	4 hours
E. One or more actuation devices inoperable.	E.1 Restore actuation device to OPERABLE status.	48 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR Minimum functional capability specified in Table 3.3.1-1 not maintained.	F.1 Enter the Condition referenced in Table 3.3.1-1.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. As required by Required Action F.1 and referenced in Table 3.3.1-1.	G.1	Reduce THERMAL POWER to < 70% RTP.	2 hours
H. As required by Required Action F.1 and referenced in Table 3.3.1-1.	H.1	Reduce THERMAL POWER to < 10% RTP.	6 hours
I. As required by Required Action F.1 and referenced in Table 3.3.1-1.	I.1	Be in MODE 2.	6 hours
J. As required by Required Action F.1 and referenced in Table 3.3.1-1.	J.1	Be in MODE 3.	6 hours
K. As required by Required Action F.1 and referenced in Table 3.3.1-1.	K.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	K.2	Open the reactor trip breakers.	6 hours
L. As required by Required Action F.1 and referenced in Table 3.3.1-1.	L.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	L.2	Reduce pressurizer pressure to < 2005 psia.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
M. As required by Required Action F.1 and referenced in Table 3.3.1-1.	M.1	Be in MODE 3.	6 hours
	AND		
	M.2	Be in MODE 4.	12 hours
N. As required by Required Action F.1 and referenced in Table 3.3.1-1.	N.1	Be in MODE 3.	6 hours
	AND		
	N.2	Be in MODE 5.	36 hours
O. As required by Required Action F.1 and referenced in Table 3.3.1-1.	O.1	Declare associated EDG inoperable.	Immediately
P. As required by Required Action F.1 and referenced in Table 3.3.1-1.	P.1	Declare associated Chemical and Volume Control System isolation valve(s) inoperable.	Immediately
Q. As required by Required Action F.1 and referenced in Table 3.3.1-1.	Q.1	Declare associated Pressurizer Safety Relief Valve(s) inoperable.	Immediately
R. As required by Required Action F.1 and referenced in Table 3.3.1-1.	R.1	Declare both Control Room Emergency Filtration trains inoperable.	Immediately
S. As required by Required Action F.1 and referenced in Table 3.3.1-1.	S.1	Open reactor trip breakers.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
T. As required by Required Action F.1 and referenced in Table 3.3.1-1.	T.1 Declare associated Actuation Logic Units inoperable. <u>AND</u> T.2 Open reactor trip breakers.	Immediately 1 hour

SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.1-1 to determine which SRs apply for each sensor, manual actuation switch, signal processor, or actuation device.

2. When a sensor, manual actuation switch, signal processor, or actuation device is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Trip/Actuation Function maintains functional capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 20% RTP. --- Compare results of calorimetric heat balance calculation to power range division output. Adjust power range division output if calorimetric heat balance calculations results exceed power range division output by more than +2% RTP.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	-------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 20% RTP. --- Perform CALIBRATION.	 15 effective full power days
SR 3.3.1.3	Perform ACTUATION DEVICE OPERATIONAL TEST.	31 days
SR 3.3.1.4	Perform CALIBRATION consistent with Specification 5.5.18, “Setpoint Control Program (SCP).”	92 days
SR 3.3.1.5	Perform a SENSOR OPERATIONAL TEST.	24 months
SR 3.3.1.6	------------------------NOTE-------------------------- Neutron detectors are excluded from CALIBRATION. --- Perform a CALIBRATION consistent with Specification 5.5.18, “Setpoint Control Program (SCP).”	24 months
SR 3.3.1.7	Perform EXTENDED SELF TESTS.	24 months
SR 3.3.1.8	Perform ACTUATION DEVICE OPERATIONAL TEST.	24 months
SR 3.3.1.9	Verify setpoints properly loaded in APUs.	24 months

Table 3.3.1-1 (page 1 of 3)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUMUM REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
A. Sensors					
1. 6.9 kV Bus Voltage	3 per EDG	1,2,3,4,(a)	2 per EDG	O	SR 3.3.1.5 SR 3.3.1.6
2. Boron Concentration - Chemical and Volume Control System (CVCS) Charging Line	4	3(b),4(b),5,6	2	P	SR 3.3.1.4 SR 3.3.1.5
3. Boron Temperature - CVCS Charging Line	4	3(b),4(b),5,6	2	P	SR 3.3.1.5 SR 3.3.1.6
4. CVCS Charging Line Flow	4	3(b),4(b),5(b)	2	P	SR 3.3.1.5 SR 3.3.1.6
5. Cold Leg Temperature (Narrow Range)	4	≥ 10% RTP	3	H	SR 3.3.1.5 SR 3.3.1.6
6. Cold Leg Temperature (Wide Range)	4	1,2(c)	3	J	SR 3.3.1.5 SR 3.3.1.6
	4	3,4,5,6(b)	2	P	SR 3.3.1.5 SR 3.3.1.6
7. Containment Pressure	4 per area	1,2,3	3 per area	M	SR 3.3.1.5 SR 3.3.1.6
8. Hot Leg Pressure (Wide Range)	4	1,2,3	3	M	SR 3.3.1.5 SR 3.3.1.6
	4	(d)	2	Q	SR 3.3.1.5 SR 3.3.1.6
9. Hot Leg Temperature (Narrow Range)	4 per division, 4 divisions	1,2(c)	3 per division, 3 divisions	J	SR 3.3.1.5 SR 3.3.1.6
10. Hot Leg Temperature (Wide Range)	4	3(e)	3	M	SR 3.3.1.5 SR 3.3.1.6

(a) When associated EDG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

(b) With three or more reactor coolant pumps (RCPs) in operation.

(c) ≥ 10^{-5}% power on the intermediate range detectors.

(d) When Pressurizer Safety Relief Valves (PSRVs) are required to be OPERABLE per LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

(e) When Table 3.3.1-2, Trip/Actuation Function B.3.a is disabled.

Table 3.3.1-1 (page 2 of 3)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUMUM REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
11. Intermediate Range	4	1(f),2,3(g)	3	K	SR 3.3.1.5 SR 3.3.1.6
12. Power Range	2 per division, 4 divisions	1,2,3(g)	2 per division, 3 divisions	K	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6
13. Pressurizer Level (Narrow Range)	4	1,2,3	3	M	SR 3.3.1.5 SR 3.3.1.6
14. Pressurizer Pressure (Narrow Range)	4	1,2,3(h)	3	L	SR 3.3.1.5 SR 3.3.1.6
15. Radiation Monitor - Containment High Range	4	1,2,3,4	3	K	SR 3.3.1.5 SR 3.3.1.6
16. Radiation Monitor - Control Room HVAC Intake Activity	4	1,2,3,4	3	N	SR 3.3.1.5 SR 3.3.1.6
	4	5,6,(i)	3	R	SR 3.3.1.5 SR 3.3.1.6
17. RCP Current	3 per RCP	1,2,3	2 per RCP	M	SR 3.3.1.5 SR 3.3.1.6
18. RCP Delta P Sensors	2 per RCP	1,2,3	1 per RCP	M	SR 3.3.1.5 SR 3.3.1.6
19. RCP Speed	4	≥ 10% RTP	3	H	SR 3.3.1.5 SR 3.3.1.6
20. Reactor Coolant System (RCS) Loop Flow	4 per loop	1,2(c)	3 per loop	J	SR 3.3.1.5 SR 3.3.1.6
21. Reactor Trip Circuit Breaker Position Indication	4	1,2(g),3g)	3	M	SR 3.3.1.5 SR 3.3.1.8
22. Self-Powered Neutron Detectors	72	≥ 10% RTP	51	H	SR 3.3.1.2 SR 3.3.1.5

(c) $\geq 10^{-5}$ % power on the intermediate range detectors.

(f) ≤ 10% RTP.

(g) With the Reactor Control, Surveillance and Limitation (RCSL) System capable of withdrawing a Rod Cluster Control Assembly (RCCA) or one or more RCCAs not fully inserted.

(h) With pressurizer pressure ≥ 2005 psia.

(i) During movement of irradiated fuel assemblies.

Table 3.3.1-1 (page 3 of 3)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUMUM REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
23. Steam Generator (SG) Level (Narrow Range)	4 per SG	1,2[(j)],3[(j)]	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6
24. SG Level (Wide Range)	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6
25. SG Pressure	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6
B. Manual Actuation Switches					
1. Reactor Trip	4	1,2,3[(g)]	3	K	SR 3.3.1.8
	4	4[(g)],5[(g)]	3	S	SR 3.3.1.8
2. Safety Injection System (SIS) Actuation	4	1,2,3,4	3	N	SR 3.3.1.8
3. SG Isolation	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.8
C. Signal Processors					
1. Remote Acquisition Units (RAUs)	2 per division, 4 divisions	≥ 10% RTP	1 per division, 4 divisions	H	SR 3.3.1.5 SR 3.3.1.7
2. Acquisition and Processing Units (APUs)	5 per division, 4 divisions	Refer to Table 3.3.1-2	Refer to Table 3.3.1-2	Refer to Table 3.3.1-2	SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9
3. Actuation Logic Units (ALUs)	4 per division, 4 divisions	1,2,3,4	3 per division, 4 divisions	N	SR 3.3.1.5 SR 3.3.1.7
	4 per division, 4 divisions	5,6,(i)	3 per division, 4 divisions	T	SR 3.3.1.5 SR 3.3.1.7
D. Actuation Devices					
1. Reactor Coolant Pump Bus and Trip Breakers	2 per pump	1,2,3,4	1 per pump	N	SR 3.3.1.8
2. Reactor Trip Circuit Breakers	4	1,2,3[(g)]	3	K	SR 3.3.1.3
3. Reactor Trip Contactors	4 per set, 23 sets	1,2,3[(g)]	3 per set, 23 sets	K	SR 3.3.1.3

(g) With the RCSL capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(i) During movement of irradiated fuel assemblies.

(j) Except when all main feedwater (MFW) isolation valves are closed.

Table 3.3.1-2 (page 1 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[a]	SETTING BASIS	CONDITION
A.	Reactor Trip				
1.a.	Low Departure from Nucleate Boiling Ratio (DNBR)	≥ 10% RTP	3 divisions	(b)	H
1.b.	Low DNBR and Imbalance or Rod Drop	≥ 10% RTP	3 divisions	(b)	H
1.c.	Variable Low DNBR and Rod Drop	≥ 10% RTP	3 divisions	(b)	H
1.d.	Low DNBR - High Quality	≥ 10% RTP	3 divisions	(b)	H
1.e.	Low DNBR - High Quality and Imbalance or Rod Drop	≥ 10% RTP	3 divisions	(b)	H
2.	High Linear Power Density	≥ 10% RTP	3 divisions	(b)	H
3.	High Neutron Flux Rate of Change (Power Range)	1,2,3[c]	3 divisions	≥ 13% RTP	K
4.	High Core Power Level	1,2[d]	3 divisions	≤ 116.7% RTP	J
5.	Low Saturation Margin	1,2[d]	3 divisions	≥ 0 Btu/lb	J
6.a.	Low-Low Reactor Coolant System (RCS) Loop Flow Rate in One Loop	≥ 70% RTP	3 divisions	≥ 50% Nominal Flow	G
6.b.	Low RCS Loop Flow Rate in Two Loops	≥ 10% RTP	3 divisions	≥ 86% Nominal Flow	H
7.	Low Reactor Coolant Pump (RCP) Speed	≥ 10% RTP	3 divisions	≥ 92% Nominal Speed	H
8.	High Neutron Flux (Intermediate Range)	1[e],2,3[c]	3 divisions	≤ 25% RTP	K

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

(b) As specified in the COLR.

(c) With the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(d) ≥ 10^{-5}% power on the intermediate range detectors.

(e) ≤ 10% RTP.

Table 3.3.1-2 (page 2 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[a]	SETTING BASIS	CONDITION
9.	Low Doubling Time (Intermediate Range)	1[e],2,3[c]	3 divisions	≥ 10 Sec.	K
10.	Low Pressurizer Pressure	≥ 10% RTP	3 divisions	≥ 1950 psia	H
11.	High Pressurizer Pressure	1,2	3 divisions	≤ 2470 psia	J
12.	High Pressurizer Level	1,2	3 divisions	≤ 83% Measuring Range	J
13.	Low Hot Leg Pressure	1,2,3[c][f]	3 divisions	≥ 1895 psia	L
14.	Steam Generator (SG) Pressure Drop	1,2	3 divisions	≥ 29 psi/min; 177 psi<ss; Max 1088 psia	J
15.	Low SG Pressure	1,2,3[c][f]	3 divisions	≥ 650 psia	M
16.	High SG Pressure	1	3 divisions	≤ 1460 psia	I
17.	Low SG Level	1,2	3 divisions	≥ 3.5% Narrow Range	J
18.	High SG Level	1,2	3 divisions	≤ 80.5% Narrow Range	J
19.	High Containment Pressure	1,2	3 divisions	≤ 19.2 psia	J

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

(c) With the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(e) ≤ 10% RTP.

(f) With pressurizer pressure ≥ 2005 psia.

Table 3.3.1-2 (page 3 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[(a)]	SETTING BASIS	CONDITION
B.	ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) SIGNALS				
1.	Turbine Trip on Reactor Trip (RT)	1	3 divisions	RT for 1 sec.	I
2.a.	Main Feedwater Full Load Closure on Reactor Trip (All SGs)	1,2[(g)]	3 divisions	NA	J
2.b.	Main Feedwater Full Load Closure on High SG Level (Affected SGs)	1,2[(g)],3[(g)]	3 divisions	≤ 80.5% Narrow Range	M
2.c.	Startup and Shutdown Feedwater Isolation on SG Pressure Drop (All SGs)	1,2[(h)],3[(h)]	3 divisions	≥ 29 psi/min; 322 psi<ss; Max 943 psia	M
2.d.	Startup and Shutdown Feedwater Isolation on Low SG Pressure (All SGs)	1,2[(h)],3[(f)(h)]	3 divisions	≥ 505 psia	L
2.e.	Startup and Shutdown Feedwater Isolation on High SG Level for Period of Time (Affected SGs)	1,2[(h)],3[(h)]	3 divisions	≤ 66.5% Narrow Range for 10 sec.	M
3.a.	Safety Injection System (SIS) Actuation on Low Pressurizer Pressure	1,2,3[(j)]	3 divisions	≥ 1613 psia	L
3.b.	SIS Actuation on Low Delta Psat	3[(i)]	3 divisions	≥ 39 psia	M
4.	RCP Trip on Low Delta P across RCP with SIS Actuation	1,2,3	3 divisions	≥ 75% Nominal Pressure	M
5.	Partial Cooldown Actuation on SIS Actuation	1,2,3	3 divisions	NA	M

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

(f) With pressurizer pressure ≥ 2005 psia.

(g) Except when all MFW full load isolation valves are closed.

(h) Except when all MFW low load isolation valves are closed.

(i) When Trip/Actuation Function B.3.a is disabled.

Table 3.3.1-2 (page 4 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[(a)]	SETTING BASIS	CONDITION
6.a.	Emergency Feedwater System (EFWS) Actuation on Low-Low SG Level (All SGs)	1,2,3	3 divisions	≥ 29% Wide Range	M
6.b.	EFWS Actuation on Loss of Offsite Power (LOOP) and SIS Actuation (All SGs)	1,2	3 divisions	NA	J
6.c.	EFWS Isolation on High SG Level (Affected SG)	1,2,3	3 divisions	≤ 98% Wide Range	M
7.a.	Main Steam Relief Train (MSRT) Actuation on High SG Pressure	1,2,3	3 divisions	≤ 1460 psia	M
7.b.	MSRT Isolation on Low SG Pressure	1,2,3[(f)]	3 divisions	≥ 505 psia	L
8.a.	Main Steam Isolation Valve (MSIV) Closure on SG Pressure Drop (All SGs)	1,2,3	3 divisions	≥ 29 psi/min; 177 psi<ss; Max 1088 psia	M
8.b.	MSIV Closure on Low SG Pressure (All SGs)	1,2,3[(j)]	3 divisions	≥ 650 psia	L
9.a.	Containment Isolation (Stage 1) on High Containment Pressure	1,2,3	3 divisions	≤ 19.2 psia	M
9.b.	Containment Isolation (Stage 1) on SIS Actuation	1,2,3,4	3 divisions	NA	N
9.c.	Containment Isolation (Stage 2) on High-High Containment Pressure	1,2,3	3 divisions	≤ 38.3 psia	M
9.d.	Containment Isolation (Stage 1) on High Containment Radiation	1,2,3,4	3 divisions	≤ 100 x background	N

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

(f) With pressurizer pressure ≥ 2005 psia.

(j) Except when all MSIVs are closed.

Table 3.3.1-2 (page 5 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[(a)]	SETTING BASIS	CONDITION
10.a. Emergency Diesel Generator (EDG) Start on Degraded Grid Voltage	1,2,3,4,(k)	NA	≥ 6089 V and ≤ 6486 V; ≥ 6.5 sec. and ≤ 12 sec. w/SIS, ≥ 6.5 sec. and ≤ 300 sec. wo/SIS	NA
10.b. EDG Start on LOOP	1,2,3,4,(k)	NA	≥ 4692 V and ≤ 5085 V; ≥ 0.17 sec. and ≤ 7 sec.	NA
11.a. Chemical and Volume Control System (CVCS) Charging Line Isolation on High-High Pressurizer Level	1,2,3	3 divisions	≤ 88% Measuring Range	M
11.b. CVCS Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating)	5[(l)],6	3 divisions	(b)	P
11.c. CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions	3,4[(m)],5[(m)]	3 divisions	(b)	P

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

(b) As specified in the COLR.

(k) When associated EDG is required to be OPERABLE by LCO 3.8.2.

(l) With two or less RCPs in operation.

(m) With three or more RCPs in operation.

Table 3.3.1-2 (page 6 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[(a)]	SETTING BASIS	CONDITION
12.a. Pressurizer Safety Relief Valve (PSRV) Actuation - First Valve	(n)	3 divisions	(o)	Q
12.b. PSRV Actuation - Second Valve	(n)	3 divisions	(o)	Q
13. Control Room Heating, Ventilation, and Air Conditioning Reconfiguration to Recirculation Mode on High Intake Activity	1,2,3,4	3 divisions	≤ 3 x background	N
	5,6,(p)	3 divisions	≤ 3 x background	R

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

(n) When the PSRVs are required to be OPERABLE by LCO 3.4.11.

(o) The LTOP arming temperature is specified in the PTLR.

(p) During movement of irradiated fuel assemblies.

3.3 INSTRUMENTATION

3.3.2 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.2 The PAM instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

--NOTE--

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required division inoperable.	A.1 Restore required division to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.5.	Immediately
C. One or more Functions with two required division inoperable.	C.1 Restore one division to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

---NOTE---

This SR applies to each PAM instrumentation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CALIBRATION	24 months
SR 3.3.2.2	Perform SENSOR OPERATIONAL TEST of the Safety Information and Control System division performing the PAM functions listed in Table 3.3.2-1.	24 months

Table 3.3.2-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED NUMBER OF DIVISIONS
1.	Cold Leg Temperature (Wide Range)	1 per loop
2.	Containment Isolation Valve Position Indication	2(a)(b)
3.	Containment Pressure	2
4.	Emergency Feedwater Storage Pool Level	1 per pool
5.	Emergency Feedwater System Flow	1 per loop
6.	Extra Boration System Flow	2
7.	Hot Leg Injection Flow	1 per loop
8.	Hot Leg Pressure (Wide Range)	1 per loop
9.	Hot Leg Temperature (Wide Range)	1 per loop
10.	In-containment Refueling Water Storage Tank Level	2
11.	Incore Temperature	2 per quadrant
12.	Power Range Monitors	2
13.	Pressurizer Level	2
14.	Radiation Monitor - Containment High Range	2
15.	Radiation Monitor - Main Steam Line Activity	1 per line
16.	Source Range Monitors	2
17.	Steam Generator Level (Wide Range)	2 per SG
18.	Steam Generator Pressure	2 per SG

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication division is required for penetration flow paths with only one installed control room indication division.

3.3 INSTRUMENTATION

3.3.3 Remote Shutdown System (RSS)

LCO 3.3.3 The RSS Functions shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable.	A.1 Restore required Functions to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Verify each required control circuit and transfer switch is capable of performing the intended function.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.3.2	-------------------------------NOTE------------------------------ Neutron detectors are excluded from the CALIBRATION. -- Perform CALIBRATION for each required instrument division.	 24 months
SR 3.3.3.3	Perform SENSOR OPERATIONAL TEST of each required Safety Information and Control System division performing the Remote Shutdown System functions.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 DNB parameters for RCS pressurizer pressure, RCS average coolant temperature, and RCS total flow rate shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

---NOTE---
RCS pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DNB parameters not within limits.	A.1 Restore the DNB parameter(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify RCS pressurizer pressure is within the limits specified in the COLR.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.1.2	Verify RCS average coolant temperature is within the limits specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is within the limits specified in the COLR.	12 hours
SR 3.4.1.4	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after ≥ 90% RTP. --- Verify by precision heat balance that RCS total flow rate is within the limits specified in the COLR.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average coolant temperature (T_{avg}) shall be ≥ 568°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Tavg in one or more RCS loops not within limit.	A.1 Be in MODE 2 with keff < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS Tavg in each loop ≥ 568°F.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE------------ Required Action A.2 shall be completed whenever this Condition is entered. --------------------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. <u>AND</u> A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5 with RCS pressure < 370 psig.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE------------ Required Action C.3 shall be completed whenever this Condition is entered. --------------------------------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits. <u>AND</u> C.2 Initiate action to reduce RCS pressure to < 370 psig. <u>AND</u> C.3 Determine RCS is acceptable for continued operation.	Immediately Immediately Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-------------------------------NOTE------------------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	30 minutes

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops – MODES 1 and 2

LCO 3.4.4 Four RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCS loop not in operation.	A.1 Reduce THERMAL POWER to ≤ 60 % <u>AND</u> A.2 Restore RCS loop to operation.	15 minutes 2 hours
B. Required Action and associated Completion Time of Condition A not met <u>OR</u> Requirements of LCO not met for reasons other than Condition A.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each RCS loop is in operation.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

LCO 3.4.5 RCS loops shall be OPERABLE and in operation as follows:

a. Four RCS loops shall be OPERABLE and in operation when the Control Rod Drive Control System (CRDCS) is capable of rod withdrawal; or

b. Two RCS loops shall be OPERABLE and one in operation when the CRDCS is not capable of rod withdrawal.

--NOTE--

All reactor coolant pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)."

b. Core outlet temperature is maintained at least 10°F below saturation temperature; and

c. The CRDCS is not capable of rod withdrawal.

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One RCS loop not in operation with CRDCS capable of rod withdrawal.	C.1 Place the CRDCS in a condition incapable of rod withdrawal.	2 hours
D. Two or more required RCS loops inoperable. <u>OR</u> Two or more RCS loops not in operation with CRDCS capable of rod withdrawal. <u>OR</u> One required RCS loop not in operation with CRDCS incapable of rod withdrawal.	D.1 Place the CRDCS in a condition incapable of rod withdrawal. <u>AND</u> D.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. <u>AND</u> D.3 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	12 hours
SR 3.4.5.2	Verify steam generator secondary side water levels are ≥ 20% for required RCS loops.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.5.3	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required pump is not in operation. ---	
	Verify correct breaker alignment and indicated power are available to each required pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two RCS loops shall be OPERABLE and one RCS loop shall be in operation.

OR

Three Residual Heat Removal (RHR) loops shall be OPERABLE and two RHR loops shall be in operation.

---NOTE--
All reactor coolant pumps and LHSI pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required loop inoperable.	A.1 Initiate action to restore required loop to OPERABLE status. AND	Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
	--------------------NOTE------------------- Only required if two RHR loops are OPERABLE. ---	
	A.2 Be in MODE 5.	24 hours
B. Two or more required loops inoperable. OR Required loop(s) not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND	
	B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required RCS or RHR loops are in operation and circulating reactor coolant at a flow rate of ≥ 2200 gpm.	12 hours
SR 3.4.6.2	Verify SG secondary side water levels are ≥ 20% for required RCS loops.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.6.3	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required system is not in operation. --- Verify correct breaker alignment and indicated power are available to each required pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of at least two steam generators (SGs) shall be ≥ 20%.

---NOTES---

1. The LHSI pump may be removed from operation for ≤ 1 hour per 8 hour period provided:

    a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; and

    b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS Loops Filled.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required RHR loop inoperable. AND One RHR loop OPERABLE.	A.1	Initiate action to restore required RHR loop to OPERABLE status.	Immediately
	OR		
	A.2	Initiate action to restore required SGs secondary side water level to within limit.	Immediately
B. One or more required SGs with secondary side water level not within limit. AND One RHR loop OPERABLE.	B.1	Initiate action to restore a second RHR loop to OPERABLE status.	Immediately
	OR		
	B.2	Initiate action to restore required SGs secondary side water level to within limit.	Immediately
C. No required RHR loops OPERABLE. OR Required RHR loop not in operation.	C.1	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND		
	C.2	Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify required RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 2200 gpm.	12 hours
SR 3.4.7.2	Verify SG secondary side water level is ≥ 20% in required SGs.	12 hours
SR 3.4.7.3	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required RHR loop is not in operation. --- Verify correct breaker alignment and indicated power are available to each required LHSI pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops – MODE 5, Loops Not Filled

LCO 3.4.8 Two Residual Heat Removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately
B. No required RHR loops OPERABLE. <u>OR</u> Required RHR loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)." <u>AND</u> B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify required RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 2200 gpm.	12 hours
SR 3.4.8.2	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after required RHR loop is not in operation. -- Verify correct breaker alignment and indicated power are available to each required LHSI pump.	 7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The Pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 75%;

b. Three groups of emergency supply pressurizer heaters OPERABLE with the capacity of each group ≥ 144 kW; and

c. The Chemical Volume and Control System (CVCS) charging and auxiliary spray valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3. AND A.2 Fully insert all rods. AND A.3 Place the Control Rod Drive Control System in a condition incapable of rod withdrawal. AND A.4 Be in MODE 4.	6 hours 6 hours 6 hours 12 hours
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
-------------NOTE---------------- Separate Condition entry is allowed for each valve. ------------------------------------ C. CVCS charging valve or auxiliary spray valve inoperable.	C.1 Isolate the associated flow path.	6 hours
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ 75%.	12 hours
SR 3.4.9.2	Verify capacity of each required group of emergency supply pressurizer heaters is ≥ 144 kW.	92 days
SR 3.4.9.3	Verify the CVCS charging and auxiliary spray valves actuate to the correct position on an actual or simulated actuation signal.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Relief Valves

LCO 3.4.10 Three Pressurizer Safety Relief Valves (PSRVs) shall be OPERABLE with a lift setting of ≥ 2484.3 psig and ≤ 2585.7 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures greater than the low temperature overpressure protection arming temperature specified in the PTLR.

--NOTE--
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the PSRVs under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PSRV inoperable.	A.1 Restore PSRV to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time of Condition A not met. OR Two or more PSRVs inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any RCS cold leg temperature less than or equal to the low temperature overpressure protection arming temperature specified in the PTLR.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each PSRV is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Low Temperature Overpressure Protection (LTOP)

LCO 3.4.11 LTOP shall be OPERABLE, consisting of the following:

a. All accumulators isolated from injecting into the RCS;

b. Miniflow lines open for any medium head safety injection (MHSI) pump capable of injecting into the RCS;

c. Reactor coolant pumps (RCPs) shall not be started unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures; and

d. One of the following pressure relief capabilities:

1. The two Pressurizer Safety Relief Valves (PSRVs) which are LTOP capable with lift settings within the limits specified in the PTLR; or

2. The RCS depressurized and an RCS vent of ≥ 10.1 square inches.

--NOTE--
An accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP arming temperature specified in the PTLR,
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

--NOTE--

LCO 3.0.4.b is not applicable when entering MODE 4.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Miniflow line of any MHSI pump capable of injecting into the RCS not open.	A.1 Initiate action to open miniflow line of the MHSI pump(s) capable of injecting into the RCS.	Immediately
	OR	
	A.2 Initiate action to verify affected MHSI pump(s) with open miniflow line is incapable of injecting into the RCS.	Immediately
B. One or more accumulators not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	B.1 Isolate affected accumulator(s).	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Increase RCS cold leg temperature greater than the LTOP arming temperature specified in the PTLR.	12 hours
	OR	
	C.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One required PSRV inoperable in MODE 4.	D.1	Restore required PSRV to OPERABLE status.	72 hours
E. One required PSRV inoperable in MODE 5 or 6.	E.1	Restore required PSRV to OPERABLE status.	12 hours
F. Required Action and associated Completion Time of Condition D or E not met. <u>OR</u> Two required PSRVs inoperable. <u>OR</u> LTOP inoperable for any reasons other than Condition A, B, D, or E.	F.1	Depressurize RCS and establish RCS vent of ≥ 10.1 square inches.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify each accumulator is isolated.	12 hours
SR 3.4.11.2	Verify each MHSI pump capable of injecting into the RCS has its associated miniflow line open.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.3	Verify required RCS vent ≥ 10.1 square inches open.	12 hours for unlocked open vent valve(s) <u>AND</u> 31 days for other vent path(s)
SR 3.4.11.4	Verify RCP start limitations are met.	Once within 15 minutes prior to each start of an RCP

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 RCS Operational LEAKAGE

LCO 3.4.12 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through each steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.12.1	-------------------------------NOTES--------------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.12.2	-------------------------------NOTE------------------------------ Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through each SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.13 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the residual heat removal (RHR) flow path when in, or during the transition to or from, the RHR mode of operation.

ACTIONS

---NOTES---

1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	--------------------NOTE------------------- Each valve used to satisfy Required Action A.1 must have been verified to meet SR 3.4.13.1 and be in the reactor coolant pressure boundary or the high pressure portion of the system. --- A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve. <u>AND</u>	 4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve. <u>OR</u> A.2.2 Restore RCS PIV to within limits.	72 hours 72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	-------------------------------NOTES------------------------------ 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when operating in the RHR mode. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. --- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5.0 gpm at an RCS pressure of ≥ 2215 psig and ≤ 2255 psig.	In accordance with the Inservice Testing Program <u>AND</u> Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months <u>AND</u> Within 24 hours following valve actuation due to automatic or manual action or flow through the valve

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Leakage Detection Instrumentation

LCO 3.4.14 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump (level or discharge flow) monitor;

b. One containment atmosphere radioactivity (particulate) monitor; and

c. One containment air cooler condensate flow rate monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment sump monitor inoperable.	A.1 --------------NOTE-------------- Not required until 12 hours after establishment of steady state operation. ------------------------------------- Perform SR 3.4.12.1.	 Once per 24 hours
	<u>AND</u>	
	A.2 Restore required containment sump monitor to OPERABLE status.	30 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required containment atmosphere radioactivity monitor inoperable.	B.1.1 Analyze grab samples of the containment atmosphere.	Once per 24 hours
	<u>OR</u>	
	B.1.2 --------------NOTE-------------- Not required until 12 hours after establishment of steady state operation. ------------------------------------ Perform SR 3.4.12.1.	Once per 24 hours
	<u>AND</u>	
	B.2.1 Restore required containment atmosphere radioactivity monitor to OPERABLE status.	30 days
	<u>OR</u>	
	B.2.2 Verify containment air cooler condensate flow rate monitor is OPERABLE.	30 days
C. Required containment air cooler condensate flow rate monitor inoperable.	C.1 Analyze grab samples of the containment atmosphere.	Once per 24 hours
	<u>OR</u>	
	C.2 --------------NOTE-------------- Not required until 12 hours after establishment of steady state operation. ------------------------------------ Perform SR 3.4.12.1.	Once per 24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required containment atmosphere radioactivity monitor inoperable. AND Required containment air cooler condensate flow rate monitor inoperable.	D.1 Restore required containment atmosphere radioactivity monitor to OPERABLE status. OR D.2 Restore required containment air cooler condensate flow rate monitor to OPERABLE status.	30 days 30 days
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours
F. All required monitors inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	Perform a CHANNEL CHECK of the required containment atmosphere radioactivity monitor.	12 hours
SR 3.4.14.2	Perform a CALIBRATION of the required containment sump monitor.	24 months
SR 3.4.14.3	Perform a CALIBRATION of the required containment atmosphere radioactivity monitor.	24 months

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.4.14.4	Perform a CALIBRATION of the required containment air cooler condensate flow rate monitor.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Specific Activity

LCO 3.4.15 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	--------------------NOTE------------------- LCO 3.0.4.c is applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 1.0 μCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	-------------------NOTE------------------- LCO 3.0.4.c is applicable. ---	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > 1.0 µCi/gm.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	-----------------------------NOTE-------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 210 µCi/gm.	 7 days
SR 3.4.15.2	-----------------------------NOTE-------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.45 µCi/gm.	 14 days AND Once between 2 and 6 hours after a THERMAL POWER change of a ≥ 15% RTP within a 1 hour period

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Steam Generator (SG) Tube Integrity

LCO 3.4.16 SG tube integrity shall be maintained.

<u>AND</u>

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection. <u>AND</u> A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.16.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 RCS Loops - Test Exceptions

LCO 3.4.17 The requirements of LCO 3.4.4, "RCS Loops – MODES 1 and 2," may be suspended provided THERMAL POWER is < 5% RTP.

APPLICABILITY: MODES 1 and 2 during startup and PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open reactor trip breakers.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify THERMAL POWER is ≤ 5% RTP.	1 hour

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Four ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > 1000 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration or enrichment not within limits.	A.1 Restore boron concentration and enrichment to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce RCS pressure to ≤ 1000 psig.	6 hours 12 hours
D. Two or more accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 1236 ft^3 and ≤ 1412.6 ft^3.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 638.2 psig and ≤ 696.2 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 1700 ppm and ≤ 1900 ppm enriched boron.	31 days <u>AND</u> --------NOTE-------- Only required to be performed for affected accumulators ------------------------ Once within 6 hours after each solution volume increase of ≥ 145 gallons, that is not the result of addition from the in-containment refueling water storage tank
SR 3.5.1.5	---------------------------NOTE------------------------------- Only required to be met when RCS pressure is ≥ 2000 psig. -- Verify power is removed from each accumulator isolation valve operator.	 31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	Verify isotopic concentration of B^{10} in each accumulator is ≥ 37%.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Four ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MHSI train inoperable.	A.1 Restore MHSI train to OPERABLE status.	120 days
B. One LHSI train inoperable.	B.1 Open ECCS cold leg cross connections. AND B.2 Restore LHSI train to OPERABLE status.	72 hours 60 days
C. Two MHSI trains inoperable. OR Two LHSI trains inoperable	C.1 Restore one inoperable train to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours
E. Less than 100% of the ECCS flow equivalent to two OPERABLE ECCS trains available.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.2	Verify ECCS piping is full of water.	31 days
SR 3.5.2.3	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.4	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.5.2.5	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.6	Verify, by visual inspection, each ECCS train suction inlet from the In-Containment Refueling Water Storage Tank is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3 Two Medium Head Safety Injection (MHSI) trains shall be OPERABLE.

APPLICABILITY: MODE 4.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MHSI train inoperable.	A.1 Restore required MHSI train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two required MHSI trains inoperable.	B.1 Be in Mode 5.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	The following SRs are applicable for all required MHSI trains: SR 3.5.2.2, SR 3.5.2.3, SR 3.5.2.4, SR 3.5.2.5, and SR 3.5.2.6.	In accordance with applicable SRs

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 In-Containment Refueling Water Storage Tank (IRWST)

LCO 3.5.4 The IRWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. IRWST temperature, boron concentration, or enrichment not within limits.	A.1 Restore IRWST temperature, boron concentration, and enrichment to within limits.	8 hours
B. IRWST inoperable for reasons other than Condition A.	B.1 Restore IRWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify IRWST borated water temperature is ≥ 59°F and ≤ 122°F.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.4.2	Verify IRWST borated water volume is ≥ 500,342 gallons and ≤ 523,703 gallons.	7 days
SR 3.5.4.3	Verify IRWST boron concentration is ≥ 1700 ppm and ≤ 1900 ppm enriched boron.	7 days
SR 3.5.4.4	Verify isotopic concentration of B^{10} in the IRWST is ≥ 37%.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Extra Boration System (EBS)

LCO 3.5.5 Two EBS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or both EBS tanks inoperable due to boron concentration or enrichment not within limits.	A.1	Restore boron concentration and enrichment to within limits.	72 hours
B. One EBS train inoperable for reasons other than Condition A.	B.1	Restore EBS train to OPERABLE status.	7 days
C. Two EBS trains inoperable for reasons other than Condition A.	C.1	Restore one EBS train to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	Verify each EBS tank borated water temperature is ≥ 68°F.	24 hours
SR 3.5.5.2	Verify total EBS tank borated water volume is ≥ 2345 ft^3.	7 days
SR 3.5.5.3	Verify each EBS tank boron concentration is ≥ 7,000 ppm and ≤ 7,300 ppm enriched boron.	31 days <u>AND</u> Once within 24 hours after water or boron is added to tank <u>AND</u> Once within 24 hours after tank temperature is restored to within limit
SR 3.5.5.4	Verify each EBS train manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days
SR 3.5.5.5	Verify each EBS pump develops a flow rate ≥ 49.0 gpm and ≤ 55.4 gpm.	In accordance with the Inservice Testing Program
SR 3.5.5.6	Verify isotopic concentration of B^{10} in each EBS tank is ≥ 37%.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.5.7	Verify flow through one EBS train from the pump into the RCS.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Post Tensioning Surveillance Program.	In accordance with the Containment Post Tensioning Surveillance Program

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTES---

1. Entry and exit is permissible to perform repairs on the affected air lock components.

2. Separate Condition entry is allowed for each air lock.

3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	------------------NOTES------------------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. --	
	A.1 Verify the OPERABLE door is closed in the affected air lock. <u>AND</u>	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2 Lock the OPERABLE door closed in the affected air lock. <u>AND</u>	24 hours
	A.3 --------------NOTE-------------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ------------------------------------- Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	-------------------NOTES------------------ 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. --	
	B.1 Verify an OPERABLE door is closed in the affected air lock. <u>AND</u>	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours
	<u>AND</u>		
	B.3	--------------NOTE-------------- Air lock doors in high radiation areas may be verified locked closed by administrative means. -------------------------------------	
		Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	<u>AND</u>		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	<u>AND</u>		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	-----------------------------NOTES---------------------------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. --- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	 In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTES--

1. Penetration flow path(s), except for the Full Flow Purge flow paths, may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE------------ Only applicable to penetration flow paths with two containment isolation valves. -------------------------------- One or more penetration flow paths with one containment isolation valve inoperable for reasons other than Condition D.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. <u>AND</u>	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	-------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ------------NOTE------------ Only applicable to penetration flow paths with two or more containment isolation valves. -------------------------------- One or more penetration flow paths with two or more containment isolation valves inoperable for reasons other than Condition D.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. ------------NOTE------------ Only applicable to penetration flow paths with only one containment isolation valve and a closed system. -------------------------------- One or more penetration flow paths with one containment isolation valve inoperable.	C.1 AND	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	C.2 -------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------ Verify the affected penetration flow path is isolated.	 Once per 31 days
D. One or more penetration flow paths with one or more full purge valves not within purge valve leakage limits.	D.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. <u>AND</u>	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	D.2 -------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------ Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
	<u>AND</u>	
	D.3 Perform SR 3.6.3.6 for the resilient seal purge valves closed to comply with Required Action D.1.	Once per 92 days
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	E.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each Full Flow Purge valve is closed.	31 days
SR 3.6.3.2	Verify each Low Flow Purge valve is closed, except when the Low Flow Purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	31 days
SR 3.6.3.3	-------------------------------NOTE------------------------------ Valves and blind flanges in high radiation areas may be verified by use of administrative controls. -- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days
SR 3.6.3.4	-------------------------------NOTE------------------------------ Valves and blind flanges in high radiation areas may be verified by use of administrative means. -- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.6	Perform leakage rate testing for Full Flow Purge valves with resilient seals.	184 days AND Within 92 days after opening the valve
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -0.2 psig and ≤ 1.2 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 131°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	24 hours

3.6 CONTAINMENT SYSTEMS

3.6.6 Shield Building

LCO 3.6.6 The Shield Building shall be OPERABLE.

---NOTE---
The Shield Building envelope may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Shield building inoperable.	A.1 Restore shield building to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify annulus negative pressure is $\geq$ 0.25 inches wg.	12 hours
SR 3.6.6.2	Verify each shield building access door is closed, except when the access opening is being used for entry and exit.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.3	Verify the annulus can be drawn down to a pressure ≥ 0.25 inches wg using one Annulus Ventilation System (AVS) train in ≤ 305 seconds after a start signal.	24 months on a STAGGERED TEST BASIS for each AVS train
SR 3.6.6.4	Verify the annulus can be maintained at a pressure ≥ 0.25 inches wg by one AVS train at a flow rate of ≤ 1295 cfm.	24 months on a STAGGERED TEST BASIS for each AVS train

3.6 CONTAINMENT SYSTEMS

3.6.7 Annulus Ventilation System (AVS)

LCO 3.6.7 Two AVS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AVS train inoperable.	A.1 Restore AVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Operate each AVS train for ≥ 15 minutes with heaters operating.	31 days
SR 3.6.7.2	Perform required AVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.7.3	Verify each AVS train actuates on an actual or simulated actuation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.8 pH Adjustment

LCO 3.6.8 The pH adjustment baskets shall contain ≥ 211 ft^3 of trisodium phosphate (TSP).

APPLICABILITY: MODES 1, 2, 3, AND 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. pH adjustment baskets contain < 211 ft^3 of TSP	A.1 Restore pH adjustment baskets to ≥ 211 ft^3 of TSP.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Verify contained volume of TSP in the pH adjustment baskets is within limits.	24 months

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Two MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSSV inoperable.	A.1 Verify associated Main Steam Relief Train is OPERABLE. AND A.2 Restore MSSV to OPERABLE status.	Immediately 30 days
B. Two MSSVs inoperable.	B.1 Restore one MSSV to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition A or B not met. OR Three or more MSSVs inoperable.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. -- Verify, for each steam generator, one MSSV lift setpoint of ≥ 1416.2 psig and ≤ 1503.8 psig and other MSSV lift setpoint of ≥ 1445.3 psig and ≤ 1534.7 psig in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ± 1%.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Four MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSIVs inoperable due to one associated control line inoperable in MODE 1.	A.1 Restore control line(s) to OPERABLE status.	72 hours
B. One MSIV is inoperable in MODE 1 for reasons other than Condition A.	B.1 Restore MSIV to OPERABLE status.	8 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 2.	6 hours
D. ------------NOTE------------ Separate Condition entry is allowed for each MSIV. --------------------------------- One or more MSIVs inoperable in MODE 2 or 3.	D.1 Close MSIV. <u>AND</u> D.2 Verify MSIV is closed.	8 hours Once per 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition D not met.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	Cycle each MSIV pilot valve.	31 days
SR 3.7.2.2	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Perform a partial closure test of each MSIV.	92 days
SR 3.7.2.3	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify the isolation time of each MSIV is within limits.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.2.4	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	24 months on a STAGGERED TEST BASIS for each MSIV pilot valve

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater (MFW) Valves

LCO 3.7.3 Four MFW Full Load Isolation Valves (MFWFLIVs), MFW Full Load Control Valves (MFWFLCVs), MFW Low Load Isolation Valves (MFWLLIVs), MFW Low Load Control Valves (MFWLLCVs), MFW Very Low Load Control Valves (MFWVLLCVs), and MFW Main Isolation Valves (MFWMIVs) shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MFWs are closed.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each MFW flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more full load flow paths with one MFW valve inoperable.	A.1 Restore MFW valve to OPERABLE status.	7 days
B. One or more full load flow paths with two MFW valves inoperable.	B.1 Restore one MFW valve to OPERABLE status.	72 hours
C. One or more full load flow paths with three MFW valves inoperable.	C.1 Restore one MFW valve to OPERABLE status.	8 hours
D. One or more low load or very low load flow paths with one or more MFWLLIV, MFWLLCV, or MFWVLLCV valves inoperable.	D.1 Isolate associated flow path. <u>AND</u> D.2 Verify the flow path is isolated.	8 hours 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the isolation time of each MFWFLIV, MFWFLCV, MFWLLIV, MFWLLCV, MFWVLLCV, and MFWMIV is within limits.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify each MFWFLIV, MFWFLCV, MFWLLIV, MFWLLCV, MFWVLLCV, and MFWMIV actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.4 Main Steam Relief Trains (MSRTs)

LCO 3.7.4 Four MSRTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSRIVs inoperable due to one or both pilot valves in control line inoperable for opening. OR One or more required MSRIVs inoperable due to one pilot valve open in one or both control lines.	A.1 Restore pilot valve(s) to OPERABLE status.	30 days
B. One or two MSRIVs inoperable for opening. OR One or two MSRIVs inoperable for closing. OR One or two MSRCVs inoperable.	B.1 Verify associated main steam safety valve(s) are OPERABLE. AND B.2 Restore valve(s) to OPERABLE status.	Immediately 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Times of Condition A or B not met. <u>OR</u> Three or more required MSRIVs inoperable for opening. <u>OR</u> Three or more required MSRVs inoperable.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each MSRIV.	24 months on STAGGERED TEST BASIS for each control line
SR 3.7.4.2	Verify one complete cycle of each MSRCV.	24 months
SR 3.7.4.3	Verify each MSRIV automatically actuates on an actual or simulated steam pressure setpoints.	24 months
SR 3.7.4.4	Verify each MSRCV is automatically positioned on an actual or simulated actuation signal.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.4.5	Verify each MSRCV is automatically switched into steam generator pressure control mode on an actual or simulated MSRIV opening signal.	24 months

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Four EFW trains shall be OPERABLE.

--NOTE--
Only one EFW train is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable for two or more EFW trains inoperable when entering MODE 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One EFW train inoperable in MODE 1, 2, or 3.	A.1 Restore EFW train to OPERABLE status.	120 days
B. Two EFW trains inoperable in MODE 1, 2, or 3.	B.1 Restore one EFW train to OPERABLE status.	72 hours
C. Required Action and associated Completion Time for Condition A or B not met. <u>OR</u> Three EFW trains inoperable in MODE 1, 2, or 3.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	6 hours 12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Four of the EFW trains inoperable in MODE 1, 2, or 3.	D.1	------------NOTE---------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. -------------------------------------	
		Initiate action to restore one EFW train to OPERABLE status.	Immediately
E. One required EFW train inoperable in MODE 4.	E.1	Initiate action to restore required EFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	Verify EFW pump suction and supply header isolation valves are locked open.	31 days
SR 3.7.5.3	Cycle each EFW discharge header cross-connect valve.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.4	Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.5	Verify, on an actual or simulated actuation signal, each EFW automatic valve that is not locked, sealed, or otherwise secured in position actuates to the correct position, each EFW pump starts automatically, and flow rate is controlled within required limits and steam generator level is controlled within limits.	24 months
SR 3.7.5.6	Verify proper alignment of the required EFW flow paths by verifying flow from the EFW storage pool to its respective steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days

3.7 PLANT SYSTEMS

3.7.6 Emergency Feedwater (EFW) Storage Pools

LCO 3.7.6 Four EFW Storage Pools shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One EFW Storage Pool inoperable.	A.1 Verify the usable volume in the three remaining EFW Storage Pools is ≥ 300,000 gal. <u>AND</u> A.2 Declare associated EFW train inoperable.	Immediately Immediately
B. Two or more EFW Storage Pools inoperable. <u>OR</u> Usable volume in EFW Storage Pools < 300,000 gal.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4, without reliance on steam generator for heat removal.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the EFW Storage Pools contain a usable volume ≥ 300,000 gal.	24 hours
SR 3.7.6.2	Verify each EFW Storage Pool supply cross connect valve is locked open.	31 days

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Four CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---
Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW System.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	---------------NOTE------------- Required Action A.1 is not applicable if CCW trains are inoperable in both CCW headers supplying the reactor coolant pump (RCP) thermal barrier cooling common loop and Condition B is entered. -------------------------------------	
		Align RCP thermal barrier cooling common loop to the CCW header with two OPERABLE CCW trains.	72 hours
	<u>AND</u>		
	A.2	Restore CCW train to OPERABLE status.	120 days
B. Two CCW trains inoperable.	B.1	Restore one CCW train to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-------------------------------NOTE------------------------------ Isolation of CCW flow to individual components, other than the RCP thermal barrier cooling common loop, does not render the CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.8 Essential Service Water (ESW) System

LCO 3.7.8 Four ESW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTES--

1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generators made inoperable by ESW System.

2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loop - MODE 4," for residual heat removal loops made inoperable by ESW System.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ESW train inoperable.	A.1 Restore ESW train to OPERABLE status.	120 days
B. Two ESW trains inoperable.	B.1 Restore one ESW train to OPERABLE status.	72 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify water level of each ESW cooling tower basin is ≥ 27.2 feet.	24 hours
SR 3.7.8.2	Verify water temperature of each ESW cooling tower basin is ≤ 90°F.	24 hours
SR 3.7.8.3	------------------------------NOTE------------------------------ Isolation of ESW flow to individual components does not render the ESW System inoperable. -- Verify each ESW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.8.4	Operate each ESW cooling tower fan for ≥ 15 minutes in all speed settings.	31 days
SR 3.7.8.5	Verify each ESW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.8.6	Verify each ESW pump and cooling tower fan starts automatically on an actual or simulated actuation signal.	24 months
SR 3.7.8.7	Verify the ability to supply makeup water to each ESW basin at ≥ 300 gpm.	24 months

3.7 PLANT SYSTEMS

3.7.9 Safety Chilled Water (SCW) System

LCO 3.7.9 Four SCW trains shall be OPERABLE and in operation.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---

Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by SCW System.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SCW train inoperable or not in operation.	A.1 Restore SCW train to OPERABLE status and in operation.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify each SCW train is in operation.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.9.2	------------------------------NOTE------------------------------ Isolation of SCW flow to individual components does not render the SCW System inoperable. --- Verify each SCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.9.3	Verify each SCW train has the capability to remove the design heat load.	24 months
SR 3.7.9.4	Verify, on an actual or simulated loss of offsite power signal, each SCW train restarts following re-energization of the associated AC electrical power division.	24 months

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Filtration (CREF)

LCO 3.7.10 Two CREF trains shall be OPERABLE.

---NOTE---
The control room envelope (CRE) may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREF train inoperable.	A.1 Restore CREF train to OPERABLE status.	7 days
B. Two CREF trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4	B.1 Initiate action to implement mitigating actions.	Immediately
	AND	
	B.2 Verify mitigating actions ensure CRE occupant exposures to radiological , and smoke hazards will not exceed limits.	24 hours
	AND	
	B.3 Restore CRE boundary to OPERABLE status.	90 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	D.1 Place OPERABLE CREF train in emergency mode. <u>OR</u> D.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
E. Two CREF trains inoperablein MODE 5 or 6, or during movement of irradiated fuel assemblies.	E.1 Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CREF trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREF train for ≥ 15 minutes with the heaters operating.	31 days
SR 3.7.10.2	Perform required CREF train testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.10.3	Verify each CREF train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

3.7 PLANT SYSTEMS

3.7.11 Control Room Air Conditioning System (CRACS)

LCO 3.7.11 Four CRACS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two CRACS trains inoperable.	A.1 Restore the inoperable CRACS train(s) to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1 Place OPERABLE CRACS train in operation. OR C.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
D. Three or more CRACS trains inoperable in MODE 5 or 6 or during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Three or more CRACS trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify each CRACS train has the capability to remove the design heat load.	24 months

3.7 PLANT SYSTEMS

3.7.12 Safeguard Building Controlled Area Ventilation System (SBVS)

LCO 3.7.12 Two SBVS Accident Exhaust Filtration trains shall be OPERABLE.

---NOTE---
The safeguards building and fuel building boundaries may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SBVS Accident Exhaust Filtration train inoperable.	A.1 Restore SBVS Accident Exhaust Filtration train to OPERABLE status.	7 days
B. Two SBVS Accident Exhaust trains inoperable due to inoperable safeguards building or fuel building boundary.	B.1 Restore safeguards building and fuel building boundaries to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Two SBVS Accident Exhaust Filtration train inoperable for reasons other than Condition B.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify safeguards building and fuel building negative pressure is ≥ 0.25 inches water gauge.	12 hours
SR 3.7.12.2	Verify each safeguards building and fuel building access door is closed, except when the access opening is being used for entry and exit.	31 days
SR 3.7.12.3	Operate each SBVS Accident Exhaust Filtration train for ≥ 15 minutes with the heaters operating.	31 days
SR 3.7.12.4	Perform required SBVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.5	Verify each SBVS Accident Exhaust Filtration train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.12.6	Verify the safeguards building and fuel building can be drawn down to a negative pressure ≥ 0.25 inches water gauge in ≤ 305 seconds after a start signal using one SBVS Accident Exhaust Filtration train.	24 months on a STAGGERED TEST BASIS for each SBVS Accident Exhaust Filtration train
SR 3.7.12.7	Verify the safeguards building and fuel building can be maintained at a negative pressure ≥ 0.25 inches water gauge using one SBVS Accident Exhaust Filtration train at a flow rate of ≤ 2640 cfm.	24 months on a STAGGERED TEST BASIS for each SBVS Accident Exhaust Filtration train

3.7 PLANT SYSTEMS

3.7.13 Safeguards Building Ventilation System Electrical Division (SBVSED)

LCO 3.7.13 Four SBVSED trains shall be OPERABLE and in operation.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SBVSED train inoperable or not in operation.	A.1 Restore SBVSED train to OPERABLE status and in operation.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify each SBVSED train is in operation.	24 hours
SR 3.7.13.2	Verify each SBVSED train has the capability to remove the design heat load.	24 months
SR 3.7.13.3	Verify, on an actual or simulated loss of offsite power signal, each SBVSED train restarts following re-energization of the associated AC electrical power division.	24 months

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Storage Pool Water Level

LCO 3.7.14 The spent fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1 --------------NOTE-------------- LCO 3.0.3 is not applicable. ------------------------------------- Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the spent fuel storage pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Storage Pool Boron Concentration

LCO 3.7.15 The spent fuel storage pool boron concentration shall be ≥ 500 ppm and boron enrichment shall be ≥ 37%.

APPLICABILITY: When fuel assemblies are stored in the spent fuel storage pool and a spent fuel storage pool verification has not been performed since the last movement of fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool boron concentration or enrichment not within limit.	--------------------NOTE------------------- LCO 3.0.3 is not applicable. ---	
	A.1 Suspend movement of fuel assemblies in the spent fuel storage pool.	Immediately
	AND	
	A.2.1 Initiate action to restore fuel storage pool boron concentration and enrichment to within limits.	Immediately
	OR	
	A.2.2 Initiate action to perform a spent fuel storage pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel storage pool boron concentration is within limit.	7 days
SR 3.7.15.2	Verify the isotopic concentration of B^{10} in the spent fuel storage pool is within limit.	24 months

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Storage

LCO 3.7.16 The combination of initial enrichment and burnup of each fuel assembly stored in Region 2 of the spent fuel storage pool shall be within the limits specified in Figure 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 of the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	--------------NOTE-------------- LCO 3.0.3 is not applicable. ------------------------------------ Initiate action to move the non-complying fuel assembly to an acceptable storage location.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1.	Prior to storing the fuel assembly in Region 2 of the spent fuel storage pool

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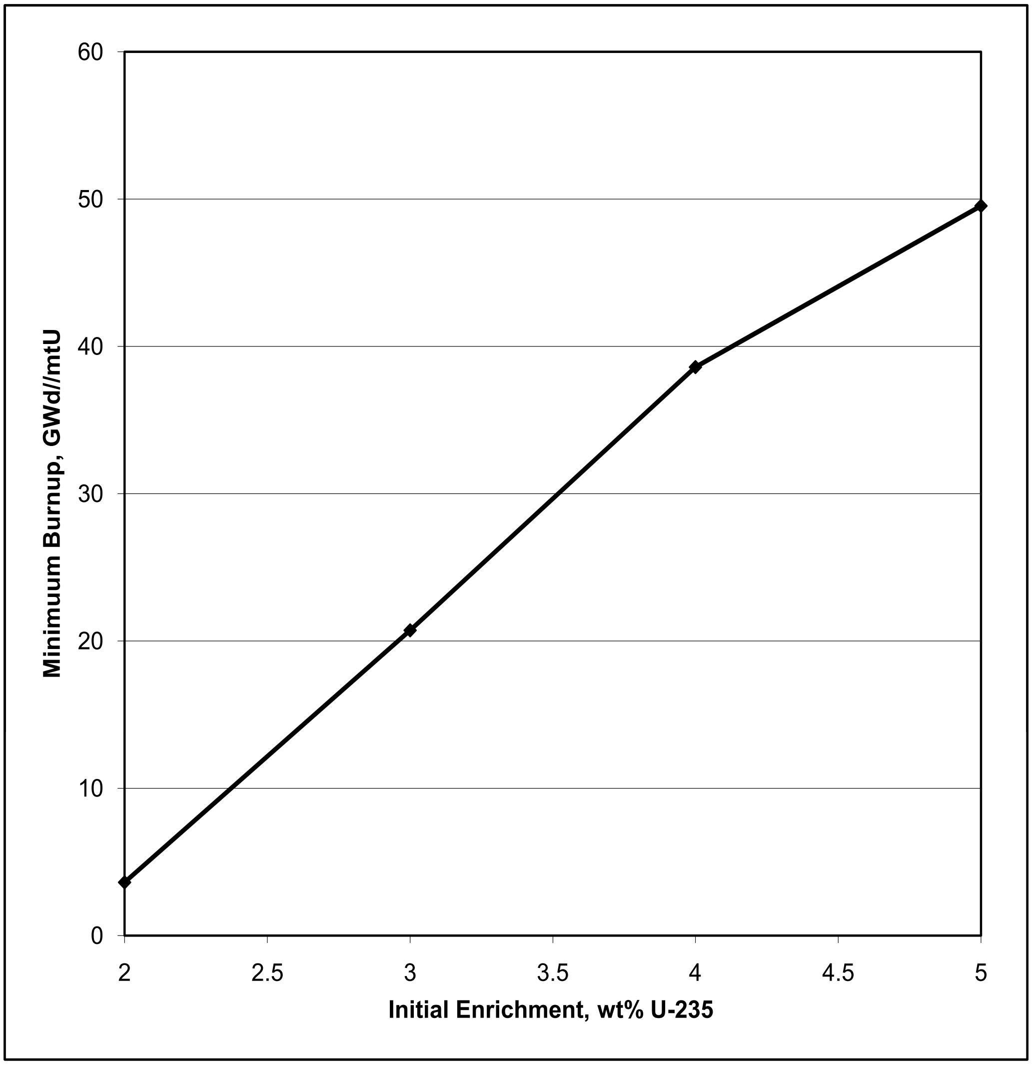

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Figure 3.7.16-1 (page 1 of 1)

Fuel Assembly Burnup Requirements for Region 2

3.7 PLANT SYSTEMS

3.7.17 Secondary Specific Activity

LCO 3.7.17 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Four emergency diesel generators (EDGs) capable of supplying the onsite Class 1E power distribution subsystems.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable with two or more EDGs inoperable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>	
	A.2 Declare required feature(s) with no offsite power available inoperable when its redundant feature(s) is inoperable.	24 hours from discovery of no offsite power to one division concurrent with inoperability of redundant feature(s)
	<u>AND</u>	
	A.3 Restore offsite circuit to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One EDG inoperable.	-------------------NOTE------------------- Required Action B.4 is not applicable if both EDGs in the same divisional pair are inoperable and Condition C is entered. ---	
	B.1 Perform SR 3.8.1.1 for the offsite circuits.	1 hour AND Once per 8 hours thereafter
	AND	
	B.2 Declare required feature(s) supported by the inoperable EDG inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND	
	B.3.1 Determine OPERABLE EDGs are not inoperable due to common cause failure.	24 hours
	OR	
	B.3.2 Perform SR 3.8.1.2 for OPERABLE EDGs.	24 hours
	AND	
	B.4 Align the alternate feed from the remaining OPERABLE EDG in the divisional pair.	72 hours
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.5 Restore EDG to OPERABLE status.	120 days
C. Two EDGs inoperable.	-------------------NOTE-------------------- Required Action C.1 is not applicable if both EDGs in the same divisional pair are inoperable. --	
	C.1 Align the alternate feed from the remaining OPERABLE EDG in one divisional pair.	2 hours
	AND	
	C.2 Restore one EDG to OPERABLE status.	72 hours
D. Two offsite circuits inoperable.	D.1 Declare required feature(s) inoperable when its redundant feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant features
	AND	
	D.2 Restore one offsite circuit to OPERABLE status.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One offsite circuit inoperable. AND One or two EDGs inoperable.	-------------------NOTE-------------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," when condition E is entered with no AC power source to any division. -- E.1 Restore offsite circuit to OPERABLE status. OR E.2 Restore all EDGs to OPERABLE status.	 12 hours 12 hours
F. Three or more EDGs inoperable.	F.1 Restore to at least two OPERABLE EDGs.	2 hours
G. Required Action and associated Completion Time of Condition A, B, C, D, E, or F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 5.	6 hours 36 hours
H. Two offsite circuits and two or more EDGs inoperable. OR One offsite circuit and three or more EDGs inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days
SR 3.8.1.2	------------------------------NOTES------------------------------ 1. All EDG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 2. A modified EDG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met. --- Verify each EDG starts from standby conditions and achieves steady state voltage ≥ 6210 V and ≤ 7260 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	31 days
SR 3.8.1.3	------------------------------NOTES------------------------------ 1. EDG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one EDG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.7. --- Verify each EDG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 8550 kW and ≤ 9500 kW.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.4	Verify each day tank contains ≥ 1350 gal of fuel oil.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	31 days
SR 3.8.1.6	Verify each fuel oil transfer system operates to automatically transfer fuel oil from storage tank to the day tank.	92 days
SR 3.8.1.7	-----------------------------NOTE------------------------------- All EDG starts may be preceded by an engine prelube period. --- Verify each EDG starts from standby condition and achieves: a. In ≤ 15 seconds, voltage ≥ 6210 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 6210 V and ≤ 7260 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	184 days
SR 3.8.1.8	Verify automatic and manual transfer of AC power sources from the normal offsite circuit to the alternate offsite circuit.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	------------------------------NOTE------------------------------ If performed with the EDG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. -- Verify each EDG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 64.5 Hz; b. Within 3 seconds following load rejection, the voltage is ≥ 6210 V and ≤ 7260 V; and c. Within 3 seconds following load rejection, the frequency is ≥ 58.8 Hz and ≤ 61.2 Hz.	24 months
SR 3.8.1.10	-----------------------------NOTE------------------------------ If performed with EDG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. -- Verify each EDG does not trip and voltage is maintained ≤ 8280 V during and following a load rejection of ≥ 8550 kW and ≤ 9500 kW.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	-----------------------------NOTES---------------------------- 1. All EDG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---	
	Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. Each EDG auto-starts from standby condition, and: 1. Energizes permanently connected loads in ≤ 15 seconds; 2. Energizes auto-connected shutdown loads through the Protection System; 3. Maintains steady state voltage ≥ 6210 V and ≤ 7260 V; 4. Maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz; and 5. Supplies permanently connected and auto-connected shutdown loads for ≥ 5 minutes.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	------------------------------NOTE------------------------------ All EDG starts may be preceded by prelube period. --- Verify on an actual or simulated Safety Injection System actuation signal each EDG auto-starts from standby condition and: a. In ≤ 15 seconds after auto-start and during tests, achieves voltage ≥ 6210 V and frequency ≥ 58.8 Hz; b. Achieves steady state voltage ≥ 6210 V and ≤ 7260 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz; c. Operates for ≥ 5 minutes; and d. Permanently connected loads remain energized from the offsite power system.	24 months
SR 3.8.1.13	Verify each EDG's noncritical automatic trips are bypassed on an actual or simulated Loss of Offsite Power signal on the emergency bus concurrent with an actual or simulated Safety Injection System actuation signal.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	------------------------------NOTES------------------------------ 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. If performed with EDG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. -- Verify each EDG operates for ≥ 24 hours: a. For ≥ 2 hours loaded ≥ 9975 kW and ≤ 10,450 kW; and b. For the remaining hours of the test loaded ≥ 8550 kW and ≤ 9500 kW.	24 months
SR 3.8.1.15	------------------------------NOTES------------------------------ 1. This Surveillance shall be performed within 5 minutes of shutting down the EDG after the EDG has operated ≥ 2 hours loaded ≥ 8550 kW and ≤ 9500 kW. Momentary transients outside of load range do not invalidate this test. 2. All EDG starts may be preceded by an engine prelube period. -- Verify each EDG starts and achieves: a. In ≤ 15 seconds, voltage ≥ 6210 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 6210 V, and ≤ 7260 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.16	-----------------------------NOTE------------------------------- This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. -- Verify each EDG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	24 months
SR 3.8.1.17	-----------------------------NOTE------------------------------- This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. -- Verify, with a EDG operating in test mode and connected to its bus, an actual or simulated Safety Injection System actuation signal overrides the test mode by: a. Returning EDG to ready-to-load operation; and b. Automatically energizing the emergency loads from offsite power.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.18	-----------------------------NOTES----------------------------- 1. All EDG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---	
	Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated Safety Injection System actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. Each EDG auto-starts from standby condition; and: 1. Energizes permanently connected loads in $\leq$ 15 seconds; 2. Energizes auto-connected emergency loads through the Protection System; 3. Achieves steady state voltage $\geq$ 6210 V and $\leq$ 7260 V; 4. Achieves steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz; and 5. Supplies permanently connected and auto-connected emergency loads for $\geq$ 5 minutes.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.19	-----------------------------NOTE------------------------------- All EDG starts may be preceded by an engine prelube period. --- Verify when started simultaneously from standby condition, each EDG achieves: a. In ≤ 15 seconds, voltage ≥ 6210 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 6210 V and ≤ 7260 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	10 years

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown"; and

b. Two emergency diesel generators (EDGs) in one divisional pair capable of supplying the onsite Class 1E power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	-------------------NOTE-------------------- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required division de-energized as a result of Condition A. --	
	A.1 Declare affected feature(s) with no offsite power available inoperable. <u>OR</u>	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. One or two required EDGs inoperable.	B.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	B.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	B.3 Initiate action to restore required EDGs to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	-----------------------------NOTE------------------------------ The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.9 through SR 3.8.1.11, SR 3.8.1.13 through SR 3.8.1.16. -- For AC sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources - Operating," except SR 3.8.1.8, SR 3.8.1.12, SR 3.8.1.17, SR 3.8.1.18, and SR 3.8.1.19 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each emergency diesel generator (EDG).

APPLICABILITY: When associated EDG is required to be OPERABLE.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each EDG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more EDGs with fuel level < 55,000 gal and > 47,000 gal in storage tank.	A.1 Restore fuel oil level to within limits.	48 hours
B. One or more EDGs with lube oil inventory < 750 gal and > 635 gal.	B.1 Restore lube oil inventory to within limits.	48 hours
C. One or more EDGs with stored fuel oil total particulates not within limit.	C.1 Restore fuel oil total particulates to within limits.	7 days
D. One or more EDGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more EDGs with starting air receiver pressure < 435 psig and ≥ 220 psig.	E.1 Restore starting air receiver pressure to ≥ 435 psig.	48 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR One or more EDGs with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated EDG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ 55,000 gal of fuel.	31 days
SR 3.8.3.2	Verify lubricating oil inventory is ≥ 750 gal.	31 days
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.4	Verify each EDG air start receiver pressure is ≥ 435 psig.	31 days
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	92 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 Four DC electrical power divisions shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required battery charger inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	2 hours
	AND	
	A.2 Verify battery float current ≤ 2 amps.	Once per 12 hours
	AND	
	A.3 Restore required battery charger to OPERABLE status.	72 hours
B. One battery inoperable.	B.1 Restore battery to OPERABLE status.	2 hours
C. One DC electrical power subsystem inoperable for reasons other than Condition A or B.	C.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and Associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	7 days
SR 3.8.4.2	Verify each required battery charger supplies ≥ 400 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours. OR Verify each required battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.3	-----------------------------NOTES------------------------------ 1. The modified performance discharge test in SR 3.8.6.6 may be performed in lieu of SR 3.8.4.3. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	24 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 Class 1E DC subsystem(s) shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required battery charger inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage. <u>AND</u> A.2 Verify battery float current ≤ 2 amps. <u>AND</u> A.3 Restore required battery charger to OPERABLE status.	2 hours Once per 12 hours 72 hours
B. One or more required DC electrical power subsystems inoperable.	B.1 Declare affected required feature(s) inoperable. <u>OR</u>	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	B.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	B.2.3 Initiate action to restore required DC subsystem(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-------------------------------NOTE------------------------------ The following SRs are not required to be performed: SR 3.8.4.2 and SR 3.8.4.3. ---	
	For DC subsystems required to be OPERABLE, the SRs of Specification 3.8.4, "DC Sources - Operating", are applicable:	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Parameters

LCO 3.8.6 Battery parameters for Divisions 1, 2, 3, and 4 batteries shall be within limits.

APPLICABILITY: When associated Class 1E DC subsystems are required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One battery with one or more battery cells float voltage < 2.07 V.	A.1 Perform SR 3.8.4.1.	2 hours
	<u>AND</u>	
	A.2 Perform SR 3.8.6.1.	2 hours
	<u>AND</u>	
	A.3 Restore affected cell voltage ≥ 2.07 V.	24 hours
B. One battery with float current > 2 amps.	B.1 Perform SR 3.8.4.1.	2 hours
	<u>AND</u>	
	B.2 Restore battery float current to ≤ 2 amps.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
---------------NOTE-------------- Required Action C.2 shall be completed if electrolyte level was below the top of plates. -------------------------------------	-------------------NOTE-------------------- Required Actions C.1 and C.2 are only applicable if electrolyte level was below the top of plates. --	
C. One battery with one or more cells electrolyte level less than minimum established design limits.	C.1 Restore electrolyte level to above top of plates.	8 hours
	<u>AND</u>	
	C.2 Verify no evidence of leakage.	12 hours
	<u>AND</u>	
	C.3 Restore electrolyte level to greater than or equal to minimum established design limits.	31 days
D. One battery with pilot cell electrolyte temperature less than minimum established design limits.	D.1 Restore battery pilot cell temperature to greater than or equal to minimum established design limits.	12 hours
E. One battery in two or more divisions with battery parameters not within limits.	E.1 Restore battery parameters for batteries in all but one division to within limits.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. <u>OR</u> One battery with one or more battery cells float voltage < 2.07 V and float current > 2 amps.	F.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	-------------------------------NOTE------------------------------ Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. --- Verify each battery float current is ≤ 2 amps.	7 days
SR 3.8.6.2	Verify each battery pilot cell float voltage is ≥ 2.07 V.	31 days
SR 3.8.6.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	31 days
SR 3.8.6.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.5	Verify each battery connected cell float voltage is ≥ 2.07 V.	92 days
SR 3.8.6.6	-------------------------------NOTE------------------------------ This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --- Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	 60 months AND 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 Divisions 1, 2, 3, and 4 inverters shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One inverter inoperable.	A.1 ---------------NOTE--------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," with any AC vital bus de-energized. ------------------------------------- Restore inverter to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to required AC vital buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 Inverters shall be OPERABLE to support the onsite Class 1E AC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to restore required inverter(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct inverter voltage, frequency, and alignments to required AC vital buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 Divisions 1, 2, 3, and 4 AC, DC, and AC Vital electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystems inoperable.	-------------------NOTE-------------------- Enter applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," for DC trains made inoperable by inoperable power distribution subsystems. --	
	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours
B. One or more AC vital electrical power distribution subsystems inoperable.	B.1 Restore AC vital electrical power distribution subsystem(s) to OPERABLE status.	2 hours
C. One or more DC electrical power distribution subsystems inoperable.	C.1 Restore DC electrical power distribution subsystem(s) to OPERABLE status.	2 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or more electrical power distribution subsystems inoperable that results in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portions of the AC, DC, and AC vital electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

--NOTE--

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or AC vital electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.3 Initiate actions to restore required AC, DC, and AC vital electrical power distribution subsystem(s) to OPERABLE status. <u>AND</u> A.2.4 Declare associated required residual heat removal loop(s) inoperable and not in operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital electrical power distribution subsystems.	7 days

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6

-----------------------------NOTE-----------------------------
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend positive reactivity additions.	Immediately
	AND	
	A.2 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1 Suspend positive reactivity additions, except the introduction of coolant into the Reactor Coolant System (RCS).	Immediately
	AND	
	A.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1, "Boron Concentration."	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.2.2	-------------------------------NOTE------------------------------ Neutron detectors are excluded from CALIBRATION. --- Perform CALIBRATION.	24 months

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch is closed and held in place by four bolts;

b. One door in each air lock is closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere is either:

   1. Closed by a manual or automatic isolation valve, blind flange, or equivalent; or

   2. Capable of being closed by an OPERABLE Containment Ventilation System.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	-------------------------------NOTE------------------------------ Not required to be met for Containment Ventilation System valve(s) in penetrations closed to comply with LCO 3.9.3.c.1. --- Verify each required Containment Ventilation System valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.9 REFUELING OPERATIONS

3.9.4 Residual Heat Removal (RHR) Loops - High Water Level

LCO 3.9.4 One RHR loop shall be OPERABLE and in operation.

---NOTE---
The required OPERABLE RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1, "Boron Concentration."

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.3 Initiate action to satisfy the requirements of the LCO. <u>AND</u>	Immediately
	A.4 Close equipment hatch and secure with four bolts. <u>AND</u>	4 hours
	A.5 Close one door in each air lock. <u>AND</u>	4 hours
	A.6 Verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by an OPERABLE Containment Ventilation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 2200 gpm.	12 hours

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) Loops - Low Water Level

LCO 3.9.5 Two Residual Heat Removal (RHR) loops shall be OPERABLE, and one RHR loop shall be in operation.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore the required number of RHR loops to OPERABLE status.	Immediately
	<u>OR</u>	
	A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No RHR loops in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1, "Boron Concentration."	Immediately
	<u>AND</u>	
	B.2 Initiate action to restore required RHR loop to operation.	Immediately
	<u>AND</u>	
	B.3 Close equipment hatch and secure with four bolts.	4 hours
	<u>AND</u>	
	B.4 Close one door in each air lock.	4 hours
	<u>AND</u>	
	B.5 Verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by an OPERABLE Containment Ventilation System	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 2200 gpm.	12 hours
SR 3.9.5.2	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required RHR loop not in operation. --- Verify correct breaker alignment and indicated power are available to each required LHSI pump.	7 days

3.9 REFUELING OPERATIONS

3.9.6 Refueling Cavity Water Level

LCO 3.9.6 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

4.0 DESIGN FEATURES

4.1 Site Location

The site for the Nine Mile Point 3 Nuclear Power Plant (NMP3NPP) is located in the north sector of Oswego County and is along the south shore of Lake Ontario, about 6 miles east of Oswego, NY. The metropolitan centers closest to the NMP3NPP site are Syracuse, NY, approximately 35 miles (56 km) to the south; Rochester, NY, approximately 70 miles (113 km) to the southwest; Buffalo, NY, approximately 140 miles (225 km) to the southwest; Scranton, PA, approximately 170 miles (274 km) to the south; Albany, NY, approximately 170 miles (274 km) to the southeast; and Toronto, Ontario, Canada, approximately 235 miles (378 km) to the west. The exclusion area boundary for NMP3NPP is a circle with a radius of 2,220 feet, or approximately 0.42 mi except for the adjacent Ontario Bible Camp property that is excluded from the EAB.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies. Each assembly shall consist of a matrix of fuel rods clad with a zirconium based alloy with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 89 control rod assemblies. The control material shall be silver indium cadmium as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in FSAR Section 9.1;

4.3 Fuel Storage (continued)

c. A nominal 10.9 inch center to center distance between fuel assemblies placed in Region 1 and a nominal 9.028 inch center to center distance between fuel assemblies placed in Region 2 of the spent fuel storage racks;

d. New or partially spent fuel assemblies with any discharge burnup may be allowed unrestricted storage in Region 1 of Figure 4.3-1;

e. Partially spent fuel assemblies meeting the initial enrichment and burnup requirements of LCO 3.7.16, “Spent Fuel Storage,” may be stored in Region 2 of Figure 4.3-1.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in FSAR Section 9.1;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in FSAR Section 9.1; and

d. A nominal 10.9 inch center to center distance between fuel assemblies placed in the new fuel storage racks.

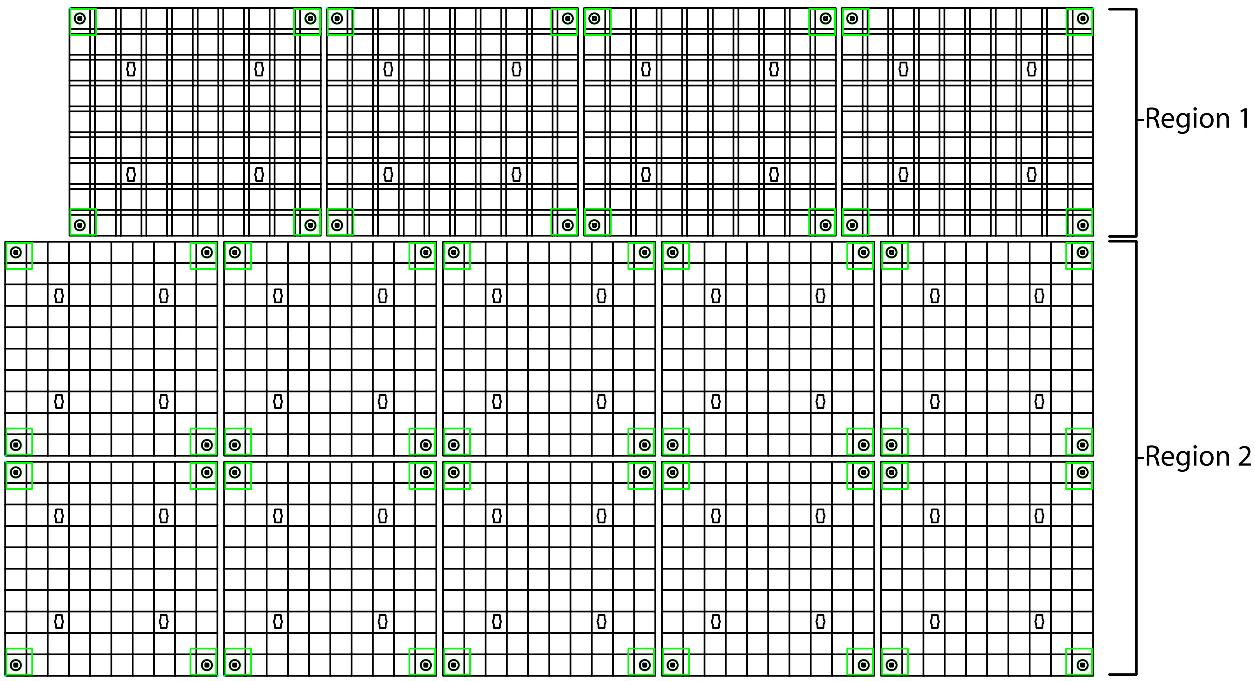
4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 23 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1360 fuel assemblies.

[



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Region 1 (Racks 1 through 4) – 360 locations

Region 2 (Racks 5 through 15) – 1000 locations

Total Storage Locations – 1360

Figure 4.3-1 (page 1 of 1)

Discrete Two Region Spent Fuel Pool Rack Layout

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

---NOTE---
Organizational positions listed or described in the Administrative Controls section shall have corresponding plant-specific titles specified in the Final Safety Analysis Report.

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 The shift supervisor shall be responsible for the control room command function. During any absence of the shift supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Operator license shall be designated to assume the control room command function. During any absence of the shift supervisor from the control room while the unit is in MODE 5 or 6, an individual with an active Senior Operator license or Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the FSAR/QA Plan.

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODE 1, 2, 3, or 4;

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements;

5.2 Organization

5.2.2 Unit Staff (continued)

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position;

d. Administrative controls shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

Any deviation from the above guidelines shall be authorized in advance by the plant manager or designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned;

e. The operations manager or assistant operations manager shall hold a Senior Operator license; and

f. When the reactor is operating in MODE 1, 2, 3, or 4, an individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 3, 2000, with the following exception:

a. During cold license operator training prior to Commercial Operation, the following Regulatory Position C.1.b of Regulatory Guide 1.8, Revision 2, 1987, applies:

   Cold license operator candidates meet the training elements defined in ANS/ANSI 3.1-1993 but are exempt from the experience requirements defined in ANS/ANSI 3.1-1993.

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed Operator are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program, and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.1 and Specification 5.6.2.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s); and

   b) A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after the approval of the plant manager; and

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Low Head Safety Injection, Medium Head Safety Injection, and Nuclear Sampling. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system once per 24 months.

The provisions of SR 3.0.2 are applicable.

5.5.3 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

5.5 Programs and Manuals

5.5.3 Radioactive Effluent Controls Program (continued)

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

1. For noble gases: a dose rate $\leq$ 500 mrem/yr to the whole body and a dose rate $\leq$ 3000 mrem/yr to the skin; and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate $\leq$ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance Frequencies.

5.5.4 Component Cyclic or Transient Limit

This program provides controls to track the FSAR Section 3.9.1.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5 Programs and Manuals

5.5.5 Containment Post Tensioning Surveillance Program

This program provides for the monitoring of the containment post tensioning force over time. Tendons used in the containment structure are fully grouted and the structure itself is not exposed to the environment during its operational life. The program shall include initial base line measurements prior to initial operation. The Containment Post Tensioning Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section III Division 2 of the ASME Boiler and Pressure Vessel Code, 2004, and Regulatory Guide 1.90, Rev. 1. Since the U.S. EPR has no ungrouted tendons, force monitoring of ungrouted tendons is not required.

The provisions of SR 3.0.3 are applicable to the Containment Post Tensioning Surveillance Program inspection frequencies.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

5.5 Programs and Manuals

5.5.7 Inservice Testing Program (continued)

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.12, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain indications with a depth equal to or exceeding 40% of the nominal tube wall thickness per eddy current results shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of Specifications 5.5.8.d.1, 5.5.8.d.2, and 5.5.8.d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems in accordance with Regulatory Guide 1.52, Revision 3, ASME N510-1989, and ASME AG-1-2003. The frequencies of 5.5.10a, 5.5.10b and 5.5.10c are in accordance with Regulatory Guide 1.52, Revision 3. The frequency for 5.5.10d and 5.5.10e is 24 months.

a. Demonstrate for each of the ESF systems that an in place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
Annulus Ventilation System (AVS)	≥ 1060 and ≤ 1295
Safeguards Building Controlled Area Ventilation System (SBCAVS)	≥ 2160 and ≤ 2640
Control Room Emergency Filtration (CREF)	≥ 3600 and ≤ 4400
Containment Low Flow Purge Subsystem (CLFPS)	≥ 2700 and ≤ 3300

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
AVS	≥ 1060 and ≤ 1295
SBCAVS	≥ 2160 and ≤ 2640
CREF	≥ 3600 and ≤ 4400
CLFPS	≥ 2700 and ≤ 3300

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 3, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below.

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Penetration	RH	Face Velocity (fpm)
AVS	0.5%	≤ 70%	300
SBCAVS	0.5%	≤ 70%	375
CREF	0.5%	≤ 70%	250
CLFPS	0.5%	≤ 70%	375

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at the system flowrate specified below..

ESF Ventilation System	Delta P (in wg)	Flowrate (cfm)
AVS	7.5	≥ 1060 and ≤ 1295
SBCAVS	7.5	≥ 2160 and ≤ 2640
CREF	7.5	≥ 3600 and ≤ 4400
CLFPS	7.5	≥ 2700 and ≤ 3300

e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF Ventilation System	Wattage (kw)
AVS	≥ 4 and ≤ 8
SBCAVS	≥ 9 and ≤ 13
CREF	
Outside Air	≥ 30 and ≤ 38
Emergency Filter Bank	≥ 13 and ≤ 17
CLFPS	≥ 12 and ≤ 16

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP Test Frequencies.

5.5 Programs and Manuals

5.5.11 Gaseous Waste Processing System Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste Processing System and the quantity of radioactivity contained in gas delay beds. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Waste Processing System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in the gas delay beds is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the beds' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

    1. An API gravity or an absolute specific gravity within limits;

    2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil; and

5.5 Programs and Manuals

5.5.12 Diesel Fuel Oil Testing Program (continued)

3. A clear and bright appearance with proper color, or a water and sediment content within limits.

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in Specification 5.5.12.a, above, are within limits for ASTM 2D fuel oil, and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program Surveillance Frequencies.

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior regulatory approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.13.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

   1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

   2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

   3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

   4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

   1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

   2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

   3. A required system redundant to the support system(s) for the supported systems described in Specifications 5.5.14.b.1 and 5.5.14.b.2 above is also inoperable.

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5.5.14 Safety Function Determination Program (SFDP) (continued)

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the approved exceptions.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident is 52.0 psig. P_a is conservatively assumed to be 55 psig. The containment design pressure is 62 psig.

c. The maximum allowable containment leakage rate, La, at Pa, shall be 0.25% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

b) For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 10 psig.

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.16 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer including the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plate.

5.5.17 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filtration System (CREFS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary;

b. Requirements for maintaining CRE boundary in its design condition including configuration control and preventive maintenance;

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0;

5.5 Programs and Manuals

5.5.17 Control Room Envelope Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization MODE of operation by one train of the CREFS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary;

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in Specification 5.5.17.c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences; and

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by Specifications 5.5.17.c and 5.5.17.d, respectively.

5.5.18 Setpoint Control Program (SCP)

a. The Setpoint Control Program shall document the Limiting Trip Setpoints (LTSPs), Nominal Trip Setpoints (NTSPs) (where desired), Allowable Values (AVs), and As-Found and As-Left Tolerance Bands for each of the required Technical Specification Instrumentation Functions in Specification 3.3.1, “Protection System (PS).”

b. The analytical methods used to determine the LTSPs, NTSPs (if applicable), AVs and As-Found Tolerance and As-Left Tolerance Bands shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. ANP-10275P-A, “U.S. EPR Instrument Setpoint Methodology Topical Report,” February 2008; and

2. ANP-10287, “Incore Trip Setpoint and Transient Methodology for U.S. EPR,” November 2007.

5.5 Programs and Manuals

5.5.18 Setpoint Control Program (SCP) (continued)

c. The Setpoint Control Program shall also contain the following:

1. Provisions for evaluation of an instrumentation division to verify it is functioning as required, before return to service, if the as-found division setpoint is found to be conservative with respect to its AV, but outside its predefined As-Found Tolerance Band; and

2. Provisions for resetting an instrumentation division setpoint to a value that is within the As-Left Tolerance Band of the associated LTSP, or within the As-Left Tolerance Band of the associated NTSP (if applicable), or otherwise declaring the instrument division inoperable.

d. The Setpoint Control Program, including any revisions or supplements, shall be provided to the NRC upon issuance:

1. Prior to initial fuel load; and
2. On a frequency consistent with 10 CFR 50.71(e).

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.2 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each cycle, or prior to any remaining portion of a cycle, and shall be documented in the COLR for the following:

1. LCO 2.1.1, "Reactor Core SLs";
2. LCO 3.1.1, "Shutdown Margin (SDM)";
3. LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
4. LCO 3.1.5, "Shutdown Bank Insertion Limits";
5. LCO 3.1.6, "Control Bank Insertion Limits";
6. LCO 3.1.7, "Rod Control Cluster Assembly (RCCA) Position Indication";
7. LCO 3.1.9, "Physics Test Exceptions – MODE 2";
8. LCO 3.2.1, "Linear Power Density";
9. LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
10. LCO 3.2.3, "Departure From Nucleate Boiling Ratio (DNBR)";
11. LCO 3.2.4, "Axial Offset";
12. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Ratio (DNB) Limits"; and
13. LCO 3.9.1, "Boron Concentration During Refueling Operations."

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. ANP-10263P, "Codes and Methods Applicability Report for the U.S. EPR";

2. DOE/ET/34212-41 BAW-1810, "Urania Gadolinia: Nuclear Model Development and Critical Experiment Benchmark," April 1984;

3. EMF-96-029(P)(A) Volume 1 and 2, "Reactor Analysis Systems for PWRs," January 1997;

4. BAW-10221P-A, "NEMO-K, A Kinetics Solution in NEMO", September 1998;

5. ANSI/ANS 19.6.1-2005, "Reload Startup Physics Tests for Pressurized Water Reactors," American Nuclear Society, 2005;

6. BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code," June 2002; and

7. BAW-10163P-A, "Core Operating Limit Methods for Westinghouse Designed PWRs," June 1989.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined assuming operation at RATED THERMAL POWER such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, low temperature overpressure protection settings, and pressurizer safety relief valve lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits;"

2. LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. ANP-10283, Rev. 0, "U.S. EPR Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown."

c. The PTLR shall be provided to the applicable regulatory body upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.2, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6 Reporting Requirements

5.6.6 Containment Post Tensioning Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Containment Post Tensioning Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, "Steam Generator (SG) Program". The report shall include:

a. The scope of inspections performed on each SG;

b. Active degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;

e. Number of tubes plugged during the inspection outage for each active degradation mechanism;

f. Total number and percentage of tubes plugged to date;

g. The results of condition monitoring, including the results of tube pulls and insitu testing; and

h. The plugging percentage for all plugging in each SG.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 <u>High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation</u>

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes Specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess one of the following:

1. A radiation monitoring device that continuously displays radiation dose rates in the area;

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint;

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source of from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door, gate, or other barrier that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designees; and

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes Specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source of from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess one of the following:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint;

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where Specifications 5.7.2.d.2 and 5.7.2.d.3 above are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displaces radiation dose rates in the area.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source of from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core

BASES

BACKGROUND

GDC 10 (Ref. 1) require that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of these SLs prevent overheating of the fuel and cladding, as well as possible cladding perforation, which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Protection System (PS) and main steam safety valves (MSSVs) prevents violation of the reactor core SLs.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The PS Limiting Trip Setpoints in LCO 3.3.1, Protection System (PS) Sensors and Signal Processors, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, AO, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the PS and the main steam safety valves.

The SLs represent a design requirement for establishing the PS Limiting Trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in FSAR Section 7.2, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to define the various PS Functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the PS precludes the violation of the above criteria, additional criteria are applied to the Linear Power Density and DNB Ratio protection functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the PS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS temperature, RCS flow rate, and AO that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SLs 2.1.1.1 and 2.1.1.2 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the PS functions are specified in LCO 3.3.1, "Protection System (PS)." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SLs 2.1.1.1 or 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable. This ensures completion within 10 CFR 50.36(d)(1)(i)(a), which requires a shutdown when safety limits are violated.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. FSAR Section 7.2.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2550 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety relief valves, MSSVs, MSRTs and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety relief valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the

BASES

APPLICABLE SAFETY ANALYSES (continued)

transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Protection System setpoints (Ref. 5), together with the settings of the MSSVs and main steam relief trains (MSRTs), provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety relief valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

a. Turbine Bypass System,

b. Reactor Control System,

c. Pressurizer Level Control System, or

d. Pressurizer spray valve.

SAFETY LIMIT

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under ASME, Section B31.1 (Ref. 6) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2803 psia.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

BASES

SAFETY LIMIT VIOLATIONS	If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour. Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria" (Ref. 4). The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized. If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 100.
5. FSAR Section 7.2.
6. ASME B31.1, Standards of Pressure Piping – Power Piping, American Society of Mechanical Engineers, 2001.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.8 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.9 and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

BASES

LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

BASES

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a unit upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met;

b. A Condition exists for which the Required Actions have now been performed; or

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if

BASES

LCO 3.0.3 (continued)

remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.14, "Spent Fuel Storage Pool Water Level." LCO 3.7.14 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.14 to "Suspend movement of irradiated spent assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

BASES

LCO 3.0.4 (continued)

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

BASES

LCO 3.0.4 (continued)

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., Containment Air Temperature, Containment Pressure, and Moderator Temperature Coefficient), and may be applied to other Specifications based on NRC plant specific approval.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

BASES

LCO 3.0.4 (continued)

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

BASES

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.14, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists.

Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6. Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required.

BASES

LCO 3.0.6 (continued)

The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained.

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.9, "PHYSICS TESTS Exceptions – MODE 2," and 3.4.17, "RCS Loops – Test Exceptions," allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

BASES

LCO 3.0.7 (continued)

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(d)(2)(ii), and, as such, are appropriate for control by the licensee.

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system’s LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

BASES

LCO 3.0.8 (continued)

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.9 and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the

BASES

SR 3.0.1 (continued)

ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

An example of this process is:

Main Steam Safety Valve (MSSV) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with MSSV considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply.

BASES

SR 3.0.2 (continued)

These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations. As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

BASES

SR 3.0.3 (continued)

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on unit risk (from delaying the Surveillance as well as any unit configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including unit shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee’s Corrective Action Program.

BASES

SR 3.0.3 (continued)

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not)

BASES

SR 3.0.4 (continued)

apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, "Frequency".

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control RCCAs, assuming that the single RCCA of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod Drive Control System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod Drive Control System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the RCCA of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

When the unit is in MODE 1 or MODE 2 with the reactor critical, SDM control is ensured by operating with the shutdown banks fully withdrawn (LCO 3.1.5) and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in MODE 2 with the reactor subcritical, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control within the estimated critical condition which includes a target Control Bank D position. When the unit is in MODE 3, 4, 5, or 6, the SDM requirements are met by means of adjustments to the RCS boron concentration.

BASES

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, AOOs, and postulated accidents;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits for the following:

   - Departure from nucleate boiling ratio (DNBR)

   - Fuel centerline temperature limits for AOOs

   - Energy deposition for the rod ejection accident based on the "Interim Acceptance Criterion and Guidance for Reactivity Initiated Accidents" contained Ref. 3; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until MODE 5 is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, the SDM requirements serve to mitigate the potential for fuel damage as a result of the post trip return to power.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition;

c. Startup of an inactive reactor coolant pump (RCP); and

d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting near the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient may be terminated by either a Excore High Neutron Flux Rate of Change trip, low DNBR trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions. Therefore, SDM satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

SDM is a core design condition that can be ensured during operation through control RCCApositioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB and the boron dilution accidents (Ref. 2) are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed SRP Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the assumptions used to determine the anti-dilution PS setpoints may no longer be valid.

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

In MODES 1 and 2 SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6 "Control Bank Insertion Limits" or by operating with the shutdown banks fully withdrawn and the control within the estimated critical condition including a target Control Bank D position.

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank, extra boration system (EBS), or the In-containment Refueling Water Storage Tank (IRWST). The operator should borate with the best source available for the plant conditions.

BASES

ACTIONS (continued)

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 1500 ppm. Assuming that a value of 1000 pcm must be recovered and a boration flow rate of 45 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 27 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1000 pcm. These boration parameters of 45 gpm and 1000 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a RCCA is known to be untrippable, however, SDM verification must account for the worth of the untrippable RCCA as well as another RCCA of maximum worth.

When the unit is in MODE 2 with the reactor subcritical, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control bank within the estimated critical condition which includes a target Control Bank D position.

In MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

BASES

SURVEILLANCE REQUIREMENTS (continued)

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, August, 2007.

2. FSAR Chapter 15.

3. NUREG-0800 Section 4.2, Appendix B, March, 2007.

4 SRP Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms," July, 2000.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences (AOOs). Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power and startup operation. The periodic confirmation of core reactivity is necessary to ensure that safety analyses for AOOs and postulated accidents remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, RCCA worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SHUTDOWN MARGIN or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as RCCA height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

BASES

BACKGROUND (continued)

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. Burnable absorbers are also depleting, increasing excess reactivity. However, the fuel typically depletes at a faster rate for an overall decrease in the excess reactivity. As the fuel depletes and burnable absorbers deplete, the RCS boron concentration is adjusted to meet the requirements in the changing reactivity, maintaining a constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control RCCA withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle. Therefore, core reactivity satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the AOO and postulated accident analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. Reactivity is sometimes expressed in units of Δk/k; however, alternative units for reactivity are %Δk/k[1] and pcm (percent millirho). The conversions between these units of reactivity are shown below.

$$1\% \frac{\Delta k}{k_1 * k_2} = 0.01 \frac{\Delta k}{k_1 * k_2}$$

$$1\ pcm = 0.00001 \frac{\Delta k}{k_1 * k_2}$$

A limit on the reactivity balance of ± 1000 pcm has been established based on engineering judgment. A 1000 pcm deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1000 pcm of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

[1] %Δk/(k_1*k_2) simplifies to %Δk/k if one of the reactivity conditions is assumed to be exactly critical (i.e. k = 1.0).

BASES

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required after each refueling due to operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a postulated accident occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

BASES

ACTIONS (continued)

The required Completion Time of 7 days is adequate for preparing operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If any required Action and associated Completion time cannot be met the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then compliance with Required Action A.1 of LCO 3.1.1.1 would result in a boration to restore SDM. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including RCCA position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (AZIMUTHAL POWER IMBALANCE, AXIAL OFFSET, etc.) for prompt indication of an anomaly.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29, August, 2007.

2. FSAR Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than zero when THERMAL POWER is ≥ 50% RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles are evaluated to ensure that the MTC does not exceed the EOC limit.

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the accident and transient analyses.

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

BASES

BACKGROUND (continued)

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

FSAR Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the RCCA withdrawal transient from either zero or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches approximately 150 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions. MTC satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO	LCO 3.1.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOC; this upper bound must not be exceeded. This maximum upper limit occurs at BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded. During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met. The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive (upper) limit and the EOC negative (lower) limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.
APPLICABILITY	Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above. In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled control rod assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no postulated accidents using the MTC as an analysis assumption are initiated from these MODES.

BASES

ACTIONS

A.1

If the upper MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

As cycle burnup is increased, the RCS boron concentration generally decreases. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits at BOC are not established within 24 hours, the unit must be brought to MODE 2 with k_{eff} < 1.0 to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the lower MTC limit means that the assumptions used in safety analysis of EOC accidents that require a bounding negative MTC may not be valid. If the EOC MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

The SR is performed prior to EOC conditions to ensure that the EOC MTC limit is not exceeded.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at BOC prior to entering MODE 1 after each refueling in order to demonstrate compliance with the most positive MTC LCO. Meeting the requirements of ANS/ANSI-19.6.1-2005 prior to entering MODE 1 ensures that this LCO for the positive MTC limit are met at other reactivity or power conditions by validating the accuracy of the physics predictions over the entire cycle.

The BOC (upper) MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOC MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

SR 3.1.3.2

In similar fashion, the LCO demands that the MTC be less negative than the specified value at the EOC full power conditions. The EOC MTC measurement may be performed at any THERMAL POWER, but the results must be extrapolated to the conditions at RTP with all banks withdrawn with the expected EOC boron concentration. This is required to have a valid comparison with the LCO value. Because the RTP MTC value gradually becomes more negative with higher core exposure and a decrease in boron concentration the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. Performing the Surveillance upon reaching 2/3 of expected projected burnup minimizes the extrapolation errors while providing sufficient warning prior to reaching the EOL (lower) MTC limit. MTC values must be extrapolated and compensated to permit direct comparison to the specified MTC limits.

SR 3.1.3.2 is modified by a Note, which indicates that if the extrapolated MTC is more negative than the EOC COLR limit, the Surveillance must be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11, August, 2007.
2. FSAR Chapter 15.
3. ANSI/ANS-19.6.1-2005, Reload Startup Physics Tests For Pressurized Reactors.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Control Cluster Assembly (RCCA) Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control RCCAs is an initial assumption in all safety analyses that assume RCCA insertion upon reactor trip. Maximum RCCA misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SHUTDOWN MARGIN (SDM).

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control or shutdown RCCA to become inoperable or to become misaligned from its group. RCCA inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available RCCA worth for reactor shutdown. Therefore, RCCA alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on RCCA alignment have been established, and all RCCA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (1 step = 10 mm ~ ⅜ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Reactor Control, Surveillance, and Limitation System (RCSL).

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. If a bank of RCCAs consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. There are four control banks and three shutdown banks.

BASES

BACKGROUND (continued)

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control RCCAs are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown RCCAs and control RCCAs is indicated by two separate and independent indicators, which are the Digital RCCA Position Indication (commonly called group step counters) and the Analog RCCA Position Indication.

The Digital RCCA Position Indication counts the pulses from the Control Rod Drive Control System that moves the RCCAs. The Digital RCCA Position Indication tracks individual RCCA positions and can display the individual RCCA position or the position for each group of RCCAs. Individual RCCAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group position. The Digital RCCA Position Indication is considered highly precise (1 step = 10 mm ~ ⅜ inch). If a RCCA does not move one step for each demand pulse, the Digital RCCA Position Indication may still count the pulse and incorrectly reflect the position of the RCCA.

The Analog RCCA Position Indication provides a totally independent indication of actual RCCA position, but at a lower precision than the step counters. This indicator is based on inductive analog signals from a series of coils spaced along a hollow tube. The Analog RCCA Position Indication is capable of measuring RCCA position within at least ± 8 steps.

APPLICABLE SAFETY ANALYSES

LCO 3.1.4 is required to ensure compliance with the control RCCA alignment and insertion limits established for anticipated operational occurrences (AOOs), and postulated accident conditions.

Control RCCA misalignment events are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control RCCA inoperability or misalignment are that:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. There will be no violations of:

   1. Specified acceptable fuel design limits; or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after AOOs and postulated accidents.

Two types of misalignment are distinguished. During movement of a control RCCA group, one RCCA may stop moving, while the other RCCAs in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control RCCAs to meet the SDM requirement, with the maximum worth RCCA stuck fully withdrawn.

Analysis is performed as follows with regard to static RCCA misalignment (Ref. 3). With the control banks at their insertion limit and again at the bite position, a power distribution is created at every combination of one and two RCCAs inserted and withdrawn 20 steps from the bank position. Satisfying limits on departure from nucleate boiling ratio in these cases bounds the situation when RCCAs are misaligned from their group by 8 steps. It is assumed that no physical mechanism can cause more than two RCCAs to be misaligned at one time within the time frame between SPND CALIBRATIONS.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 3).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear power densities will not occur, and that the requirements on SDM and ejected rod worth are preserved.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Continued operation of the reactor with a misaligned control RCCA is allowed if the limits on AXIAL OFFSET (AO), AZIMUTHAL POWER IMBALANCE (API), Departure from Nucleate Boiling Ratio (DNBR), and Linear Power Density (LPD) are verified to be within their limits and the safety analysis is verified to remain valid. When a control RCCA is misaligned, the assumptions that are used to determine the rod insertion limits, AO limits, and API limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and the self powered neutron detectors (SPNDs) must be used to verify limits. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the operating limits.

Shutdown and control RCCA OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in the safety analyses. Therefore, they satisfy Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

The limits on shutdown or control RCCA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on control RCCA OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control RCCA OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and RCCA alignment. The RCCA OPERABILITY requirement is satisfied provided the RCCA will fully insert in the required RCCA drop time assumed in the safety analysis. RCCA control malfunctions that result in the inability to move a RCCA (e.g., RCCA lift coil failures), but that do not impact trippability, do not result in RCCA inoperability.

The requirement to maintain the RCCA alignment to within plus or minus 8 steps is conservative. The minimum misalignment assumed in safety analysis is 20 steps (~8 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LPDs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

BASES

APPLICABILITY

The requirements on RCCA OPERABILITY and RCCA alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of RCCAs have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control RCCAs are fully inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control RCCAs have the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

When one or more RCCAs are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

In this situation, SDM verification must include the worth of the untrippable RCCA, as well as an RCCA of maximum worth stuck out of the core.

A.2

When one or more RCCAs are inoperable the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

B.1.1 and B.1.2

With one misaligned RCCA, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned RCCA may not be desirable. For example, realigning control bank B to a RCCA that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2, B.3, B.4, and B.5

For continued operation with one misaligned RCCA, RTP must be reduced, SDM must periodically be verified within limits, LPD, DNBR, AO, and API must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to ≤50% RTP ensures that local power density increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 1). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Protection System (PS).

When a RCCA is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

BASES

ACTIONS (continued)

Performing a flux map using the Aeroball Measurement System, and calibrating the self powered neutron detectors ensures that the SPND constants are updated to reflect any localized power redistributions. This ensures that accurate monitoring of the LPD, DNBR, and AO is maintained. The Completion Time of 12 hours allows sufficient time to obtain an Aeroball Measurement Map and to incorporate calibration constants.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during an AOO or postulated for the duration of operation under these conditions. The accident analyses presented in Chapter 15 (Ref. 3) that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

When Required Actions cannot be completed within their Completion Time of Condition B, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control RCCA becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases or LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

BASES

ACTIONS (continued)

D.2

If more than one RCCA is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual RCCA positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a RCCA that is beginning to deviate from its expected position. The specified Frequency takes into account other RCCA position information that is continuously available to the operator in the control room, so that during actual RCCA motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each control RCCA is OPERABLE would require that each RCCA be tripped. However, in MODES 1 and 2 with $k_{eff} \geq 1.0$, tripping each control RCCA would result in azimuthal or axial power tilts, or oscillations. Exercising each individual control RCCA every 92 days provides increased confidence that all RCCAs continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control RCCA by 16 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the RCCAs. Between required performances of SR 3.1.4.2 (determination of control RCCA OPERABILITY by movement), if a control RCCA(s) is discovered to be immovable, but remains trippable, the control RCCA(s) is considered to be OPERABLE. At any time, if a control RCCA(s) is immovable, a determination of the trippability (OPERABILITY) of the control RCCA(s) must be made and appropriate action taken.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verification of RCCA drop times allows the operator to determine that the maximum RCCA drop time permitted is consistent with the assumed RCCA drop time used in the safety analysis. Measuring RCCA drop times prior to reactor criticality, after each reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with RCCA motion or RCCA drop time, and that no degradation in these systems has occurred that would adversely affect control RCCA motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions. Performing rod drop testing at less the temperature specified for hot zero power is conservative due to increased reactor coolant density at lower temperature and the associated increase in rod drop resistance.

This Surveillance is performed prior to criticality after each removal of the reactor head, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, August, 2007.
2. 10 CFR 50.46, August, 2007.
3. FSAR Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown bank Rod Control Cluster Assemblies (RCCAs) are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SHUTDOWN MARGIN (SDM) and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The RCCAs are divided among four control banks and three shutdown banks. Each shutdown bank may be further subdivided into groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. See LCO 3.1.4, "RCCA Group Alignment Limits," for RCCA alignment requirements, LCO 3.1.6, "Control Bank Insertion Limits," for control bank insertion limits, LCO 3.1.7 "RCCA Position Indication for position indication requirements.

The shutdown banks are fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. Since the shutdown banks are fully withdrawn, they do not affect core power and burnup distribution, but will add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

BASES

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from 100% RTP conditions at rated temperature to 0% RTP, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit ensures that the shutdown banks do not need to be considered in a control rod ejection event analysis.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

a. There are no violations of:

   1. Specified acceptable fuel design limits; or

   2. RCS pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

BASES

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In other conditions in MODE 3 and MODE 4, 5, or 6, the shutdown banks are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2, and A.2

When one or more shutdown banks are not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If any required Action and associated Completion Time is not met, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits 4 hours prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28, August, 2007.

2. 10 CFR 50.46, August, 2007.

3. FSAR Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the Control Bank Rod Cluster Control Assemblies (RCCAs) are initial assumptions in all safety analyses that assume RCCA insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SHUTDOWN MARGIN (SDM), and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on Control Bank RCCA insertion have been established, and all RCCA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected RCCA worth, reactivity insertion rate, and SDM limits are preserved.

The RCCAs are divided among four control banks and three shutdown banks. Each shutdown bank may be further subdivided into groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. See LCO 3.1.4, "RCCA Group Alignment Limits," for RCCA OPERABILITY and alignment requirements, LCO 3.1.5, "Shutdown Bank Insertion Limits" for shutdown bank insertion limits, and LCO 3.1.7, "RCCA Position Indication" for position indication requirements.

The Control Bank RCCA groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between RCCA worth and position (integral RCCA worth). The Control Bank RCCA groups are withdrawn and operate in a predetermined sequence. The control bank RCCA group sequence and overlap limits are specified in the COLR.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Reactor Control, Surveillance, and Limitation System (RCSL), but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

BASES

BACKGROUND (continued)

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.4, "RCCA Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.1.7, "RCCA Position Indication," LCO 3.2.4, "AO," and LCO 3.2.4, "API" provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

Operation within the power density limits prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling Systems analysis. Operation within the API and departure from nucleate boiling (DNB) limits prevents DNB during a loss of forced reactor coolant flow event.

In addition to the LPD, AO, API, and DNBR limits, certain reactivity limits are preserved by regulating RCCA insertion limits. The Control Bank RCCA insertion limits also restrict the ejected RCCA worth to the values assumed in the safety analyses and preserve the minimum required SDM in MODES 1 and 2.

The Control Bank RCCA insertion and alignment limits, API and AO, are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the Control Bank insertion limits control the reactivity that could be added in the event of a RCCA ejection accident and the Shutdown and Control Bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow, ejected RCCA, or other accident requiring termination by a Protection System function.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and anticipated operational occurrences (AOOs). The acceptance criteria for the Control Bank RCCA insertion, AO, and API LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F, 10 CFR 50.46 (Ref. 2);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that a DNB condition does not exist with an RCCA inserted into the core;

c. Energy deposition for the rod ejection accident based on the "Interim Acceptance Criterion and Guidance for Reactivity Initiated Accidents" contained in Standard Review Plan (SRP) Section 4.2 Appendix B (Ref. 3), and

d. The RCCAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth RCCA stuck fully withdrawn, GDC 26 (Ref. 1).

e. The core remains subcritical after accident transients.

Control Bank position, API, and AO are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local power densities.

The SDM requirement is ensured by limiting the Control and Shutdown Bank RCCA insertion limits, so that the allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the RCCAs to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 1).

The most limiting SDM requirements for MODE 1 and 2 conditions are determined by the requirements of several transients, e.g., loss of flow, seized rotor, steam line break, etc.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The measurement of RCCA bank worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adherence to LCOs 3.1.5 and 3.1.6 provides assurance that the available SDMs at any time in cycle will exceed the limiting SDM requirements at that time in cycle.

Operation at the insertion limits or AO may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed API present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The Control and Shutdown Bank RCCA insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected RCCA worth, and power distribution peaking factors are preserved (Ref. 3).

The Control Bank RCCA insertion limits satisfy Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

The limits on Control Bank RCCA sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected RCCA worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion and withdrawal, and is imposed to maintain acceptable power peaking during Control Bank RCCA motion.

APPLICABILITY

The Control Bank RCCA sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and MODE 2. These limits must be maintained, since they preserve the assumed power distribution, ejected RCCA worth, SDM, and reactivity rate insertion assumptions. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The control banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected RCCA worth assumptions would be exceeded in these MODES. SDM is preserved in MODES 3, 4, and 5 by adjustments to the soluble boron concentration.

BASES

APPLICABILITY (continued)

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. SR 3.1.4.2 verifies the freedom of the RCCAs to move, and requires the control bank RCCAs to move below the LCO limits, which would normally violate the LCO. The Note also allows the LCO to be not applicable during a partial trip, which inserts a selected RCCA group, specified in the COLR, during loss of load events. This condition is outside of the control bank insertion limits and requires prompt action to restore operation within insertion limits.

ACTIONS

A.1.1, A.1.2, and A.2

Operation beyond the insertion limits may result in a loss of SDM and excessive peaking factors. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

The insertion limits should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the RCCAs in response to changing plant conditions. When the Control Bank groups are inserted beyond the insertion limits, actions must be taken to either withdraw the Control Bank groups beyond the insertion limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual RCCA insertion limit. Verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1. The allowed Completion Time of 2 hours provides a reasonable time to restore control banks within insertion limits, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

BASES

ACTIONS (continued)

B.1.1, B.1.2, and B.2

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlaps limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

C.1

If Required Actions A.1.1, A.1.2, and A.2, or B.1.1, B.1.2, and B.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 2 with k_{eff} < 1.0, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical condition (ECC) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECC substantially in error. Conversely, determining the ECC immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECC calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECC calculation with other startup activities.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.6.2

Verification of the control bank insertion, sequence, and overlap limits is required to detect control banks that may be approaching the insertion, sequence and overlap limits. During normal operation it is unlikely that this should be a problem since normally, very little rod motion occurs in 12 hours and automatic rod motion is controlled by the Reactor Control, Surveillance and Limitation System (RCSL). The operator is responsible for monitoring each RCSL automatic movement of the control banks and to take manual actions if necessary to maintain insertion, sequence, and overlap limits. A Completion Time of 12 hours is adequate to ensure that the control banks are maintained within insertion, sequence, and overlap limits.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28, August, 2007.
2. 10 CFR 50.46, August, 2007.
3. FSAR Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Rod Control Cluster Assembly (RCCA) Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

In addition, the analog rod position indication provides input into the Low DNBR PS functions. The indication is used to detect dropped rods and initiate automatic protection setpoint adjustments to account for these off-normal conditions.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among four control banks and three shutdown banks. Each control bank may be further subdivided into two groups to provide for precise reactivity control. The axial position of shutdown rods and control rods are determined by two separate and independent indications: the Digital Rod Position Indication (commonly called group step counters) and the Analog Rod Position Indication.

BASES

BACKGROUND (continued)

The Digital RCCA Position Indication counts the pulses from the control rod drive control system that moves the RCCA s. The Digital RCCA Position Indication tracks individual RCCA positions and can display the individual RCCA position or the position for each group of RCCAs. Individual RCCAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group position. The Digital RCCA Position Indication is considered highly precise (1 step = 10 mm ~ ⅜ inch). If a RCCA does not move one step for each demand pulse, the Digital RCCA Position Indication may still count the pulse and incorrectly reflect the position of the RCCA.

The Analog RCCA Position Indication provides a totally independent indication of actual RCCA position, but at a lower precision than the step counters. This indication is based on inductive analog signals from a series of coils spaced along a hollow tube. The Analog RCCA Position Indication is capable of measuring RCCA position within at least ± 8 steps.

APPLICABLE SAFETY ANALYSES

Control and shutdown RCCA position accuracy is essential during power operation. Power peaking, ejected rod or SDM limits may be violated in the event of an AOO or postulated accident (Ref. 2), with control or shutdown RCCAs operating outside their limits undetected. Therefore, the acceptance criteria for RCCA position indication is that RCCA positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The RCCA positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "RCCA Group Alignment Limits"). Control RCCA positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

In addition, RCCA position indication is necessary to support automatic adjustment of Low DNBR protection system setpoints in the event of a sensed RCCA or multiple RCCA drops.

The control RCCA position indicator divisions satisfy Criterion 2 of 10 CFR 50.36(d)(2)(ii). The control RCCA position indicators monitor control RCCA position, which is an initial condition of the accident.

BASES

LCO

The control RCCA position indicators are considered capable of monitoring RCCA position when:

a. The Analog RCCA Position Indication is within 8 steps of the Digital RCCA Position Indication as required by this LCO.

b. For the Analog RCCA Position Indication the primary and secondary coils indicate RCCA position; and

c. The Digital RCCA Position Indication has been calibrated either in the fully inserted position or to the Analog RCCA Position Indication. The 8 step agreement limit between the Digital RCCA Position Indication and the Analog RCCA Position Indication signifies that the Digital RCCA Position Indication is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, in position indication for a single control RCCA, ensures high confidence that the position uncertainty of the corresponding control RCCA group is within the assumed values used in the analysis (that specified control RCCA group insertion limits).

These requirements ensure that control RCCA position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicator divisions ensures that inoperable, misaligned, or mispositioned control RCCAs can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the Analog RCCA Position Indication and the Digital RCCA Position Indication are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of RCCAs have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

BASES

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable RCCA position indicator and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1.1 and A.1.2

When a single RCCA position indication in one or more banks is not capable of measuring RCCA position there is no method for determining the RCCA position. This results in the PS being unable to detect if that RCCA were to inadvertently drop into the core. Operation in this condition for any extended period of time therefore requires that the DNBR trip setpoints be adjusted to account for the possibility that a RCCA may become misaligned beyond the LCO limits or drop into the core undetected. Implementation of the DNBR penalty on the PS setpoint within the Completion Time of 8 hours is adequate since the probability of experiencing a dropped or misaligned RCCA during this period is extremely small.

With the failure of the RCCA position indication, detection of a misalignment may still be done indirectly through use of the Aeroball Measurement System. Additionally, movement of RCCAs with inoperable position indicators increases the potential for creating an undetected misalignment. Based on experience, normal power operation requires little movement of control banks. Therefore a Completion Time of Once per 8 hours (and once within 4 hours when a RCCA is moved in excess of 20 steps) is adequate for allowing full power operation.

A.2

Another option for a single RCCA position indication failure is to reduce power below that which DNBR, power peaking, or xenon oscillations are concerns.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for RCCA position determination by Required Action A.1.2 above.

BASES

ACTIONS (continued)

B.1, B.2, and B.3

When more than one Analog RCCA Position Indication in one or more banks fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of RCCA misalignment on associated accident analyses are limited.

Verifying that at least three rod control cluster assembly units (RCCAUs) are OPERABLE ensures that the majority of the analog RCCA position indicators remain available to provide necessary plant information. Placing the Control Rod Drive Control System in manual assures unplanned RCCA motion will not occur. The immediate Completion Time for placing the Control Rod Drive Control System in manual reflects the urgency with which unplanned RCCA motion must be prevented while in this Condition as well as the need to maintain adequate monitoring of RCCA position.

With the RCCAs under manual control it is now possible to more effectively monitor for a misaligned or dropped RCCA based on changes in RCS T_{AVG} since only minor fluctuations are expected during steady-state operation. A Completion Time of Once per hour is sufficient identify abnormal trends.

B.4

The 24 hour Completion Time provides sufficient time to troubleshoot and restore the Analog RCCA Position Indication to operation while avoiding the plant challenges associated with the shutdown without full RCCA position indication.

C.1.1 and C.1.2

With one Digital RCCA Position Indicator per bank not capable of indicating RCCA position, the RCCA positions can be determined by the Analog RCCA Position Indication. Since normal power operation does not require excessive movement of RCCAs, verification by administrative means the RCCA positions of all affected banks that the most withdrawn RCCA and the least withdrawn RCCA are ≤ 8 steps apart within the allowed Completion Time of once every 8 hours is adequate.

BASES

ACTIONS (continued)

C.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where RCCA position is not significantly affecting core peaking factor limits (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the RCCA positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the analog RCCA position indication agrees with the digital RCCA position within 8 steps ensures that the analog RCCA position indication is operating correctly. Since the analog RCCA position indication does not display the actual RCCA position below 10 steps, only points within the indicated ranges are required in comparison.

This Surveillance is performed prior to reactor criticality after each removal of the reactor head, as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28, August, 2007.

2. 10 CFR 50.46, August, 2007.

3. FSAR Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Boron Dilution Protection (BDP)

BASES

BACKGROUND

The primary purpose of the Boron Concentration Measurement System (BCMS) is to mitigate the consequences of the inadvertent addition unborated primary grade water into the Reactor Coolant System (RCS) when the reactor is in MODES 3, 4, 5, and 6. The RCS boron concentration that is measured by the BCMS is continuously compared to one of three pre-established setpoints that is depends on the following conditions:

- Reactor critical;
- Shutdown, with RCPs in operation; and
- Shutdown, without RCPs in operation.

The BCMS setpoint ensures:

- The dilution is terminated when the Protection setpoint is actuated;
- The available SDM is sufficient to shutdown the core with the RCCA with highest worth unable to insert, if at power; and
- The core remains sub-critical if already shutdown.

The BCMS setpoint is periodically adjusted to compensate for core burnup and the indicated boron concentration is periodically compared against boron titration samples and boron isotopic analyses to confirm that the BCMS measured boron concentration is within analysis assumptions. The BCMS instrumentation requirements are specified in Technical Specification 3.3.1 "Protection System."

The volume control tank (VCT) and letdown isolation valves actuate to the isolation position on a signal from the BCMS. The OPERABILITY requirements for these isolation valves help ensure that a dilution path is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

BASES

APPLICABLE SAFETY ANALYSES

The BCMS responds to abnormal increases in neutron counts per second (flux rate) and actuates the VCT and letdown isolation valves to mitigate the consequences of an inadvertent boron dilution event as described in FSAR Chapter 15 (Ref. 1). The accident analyses rely on the VCT and letdown isolation valves to terminate a boron dilution event. The IRWST isolation valve is also sent a signal to open to protect the CVCS charging pump and provide uninterrupted flow to the RCP seals but it is not credited in the Chapter 15 accident analyses.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the VCT and letdown isolation valves. Two valves in series in the charging pump suction line provide assurance that the dilution path could be isolated even if a single failure occurred.

The BDP VCT and letdown isolation valves satisfies Criterion 3 of 10 CFR 50.36(d)(3)(ii) (Ref. 2)

LCO

LCO 3.1.8 provides the requirements for OPERABILITY of the VCT and letdown isolation valves that mitigate the consequences of an inadvertent boron dilution event as described in FSAR Chapter 15 (Ref. 1). The VCT and letdown isolation valves are as follows:

- VCT Outlet (KBA21 AA001)
- Letdown to Charging Pump Suction Header (KBA21 AA017)
- VCT and Letdown to Charging Pump Common Isolation (KBA21 AA009)

This LCO provides assurance that the VCT and letdown isolation valves will perform their designed safety functions to mitigate the consequences of inadvertent boron dilution events.

APPLICABILITY

The VCT and letdown isolation valves must be OPERABLE in MODES 3, 4, 5, and 6 because the safety analysis identifies the VCT and letdown isolation valves as the primary means to mitigate an inadvertent boron dilution of the RCS.

BASES

ACTIONS

A.1 and A.2

If one or more of the VCT and letdown isolation valves are inoperable, the automatic capability for mitigation of dilution events is no longer available. In this case, the isolation valves are required to be restored to OPERABLE status within 8 hours. As an alternative (Required Actions A.1 and A.2), the VCT and letdown return line must be closed and secured within 8 hours to isolate the unborated water sources. The allowed Completion Times are reasonable, based on operating experience, to return the isolation valves to an OPERABLE condition in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Periodic surveillance testing of VCT and letdown isolation valves is required by the ASME Code. This verifies that the measured performance, receipt of isolation signal to full closure, is within an acceptable tolerance of the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code.

The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.1.8.2

This periodic surveillance is performed on the VCT and letdown isolation valves to verify that the actuation signal causes the appropriate valves to move to their correct position within the allowable design basis response time.

The 24 month frequency is based on the need to perform this Surveillance under conditions that apply during a plant outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The Frequency is acceptable based on consideration of the design reliability of the equipment. The actuation logic is tested as part of Protection System testing.

REFERENCES

1. FSAR Chapter 15.

2. 10 CFR 50.36, Technical Specifications, August, 2007.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.52 (Ref. 2) and 10 CFR 50.59 (Ref. 3).

The key objectives of a test program are to (Ref. 4):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 5).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

BASES

BACKGROUND (continued)

The PHYSICS TESTS required for reload fuel cycles (Ref. 5) in MODE 2 are listed below:

a. Critical Boron Concentration - Control Rods Withdrawn;

b. Critical Boron Concentration - Control Rods Inserted;

c. Control Rod Worth;

d. Isothermal Temperature Coefficient (ITC); and

e. Neutron Flux Symmetry.

The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2. If the neutron flux symmetry test is performed in MODE 1 a PHYSICS TEST Exception is required, however if the test is performed in MODE 2 a PHYSICS TEST Exception is not required. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1000 pcm when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined.

BASES

BACKGROUND (continued)

The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "RCCA Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," or LCO 3.1.6, "Control Bank Insertion Limits."

c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.3, 3.1.4, LCO 3.1.5, or LCO 3.1.6.

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could challenge LCO 3.4.2, "RCS Minimum Temperature for Criticality."

BASES

BACKGROUND (continued)

e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at ≤ 30% RTP (Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The purpose of the LCOs that are excepted by this LCO are described in the Bases for the individual LCOs (Ref. 6). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The requirements for initial testing of the facility, including PHYSICS TESTS have been defined. FSAR Section 14.2 summarizes the zero, low power, and power ascension tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-2005 (Ref. 5). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.4 "RCCA Group Alignment Limits," LCO 3.1.5 "Shutdown Bank Insertion Limits," and LCO 3.1.6 "Control Bank Insertion Limits" are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is maintained ≥ 568°F, and SDM is within the limits provided in the COLR.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AXIAL OFFSET (AO) and AZIMUTHAL POWER IMBALANCE (API), which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The API limit is specified in LCO 3.2.5.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(d)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

This LCO allows the selected control and shutdown banks/rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, 3.1.4, LCO 3.1.5, and LCO 3.1.6, may be suspended during the performance of PHYSICS TESTS provided:

a. SDM is within the limits provided in the COLR; and

b. THERMAL POWER is < 5% RTP.

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enters MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

BASES

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. The 1 hour Completion Time to suspend PHYSICS TESTS exceptions provides sufficient time to restore SDM to within limit and reflects the low risk of postulated accidents at the conditions that exist during PHYSICS TESTS.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.9.2

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

BASES

SURVEILLANCE REQUIREMENTS (continued)

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration;

g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH);

h. Moderate defect, when above the POAH; and

i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI, March, 2006.

2. 10 CFR 50.52 "Licenses, Certifications, and Approvals for Nuclear Power Plants," August, 2007.

3. 10 CFR 50.59 "Changes, Tests and Experiments," August, 2007.

4. Regulatory Guide 1.68, Revision 2, August, 1978.

5. ANSI/ANS-19.6.1-2005, November 29, 2005.

6. ANP-10263P Codes and Methods Applicability Report for the U.S. EPR, August, 2006.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Linear Power Density (LPD)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss of Coolant Accident (LOCA), loss of flow accident, rod ejection accident, or other postulated accident requiring termination by a Protection System (PS) trip function. This LCO limits the damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of the transient.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., rod control cluster assembly (RCCA) insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting axial power redistribution over time also minimizes xenon distribution swings, which is a significant factor in controlling the axial power distribution.

The power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the bounding conditions for power distribution is accomplished by maintaining the local LPD and the Departure from Nucleate Boiling Ratio (DNBR) within limits.

Proximity to the Departure from Nucleate Boiling (DNB) condition is expressed by the DNBR which is defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux.

There are two systems that perform online monitoring of the core maximum LPD and minimum DNBR: the Protection System (PS) and the Reactor Control, Surveillance and Limitation (RCSL) system. The PS and the RCSL system are capable of verifying that the LPD and the DNBR do not exceed their limits. The PS and the RCSL system perform this

BASES

BACKGROUND (continued)

function by continuously monitoring incore Self-Powered Neutron Detectors (SPND) measurements, thermal-hydraulic data, and RCCA insertion, and by calculating an actual value of DNBR and LPD, for comparison to:

- the respective trip setpoints in the PS and;
- the limits for critical operation in the RCSL system.

Thus the RCSL system continuously indicates to the operator how far the core is from the operating LPD and DNBR limits, and provides an alarm in the Main Control Room if any limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an AOO or postulated accident, but does not necessarily imply a violation of fuel design limits.

The calculation of the maximum linear power density (in kW/ft) in the PS (High Linear Power Density function) and in the RCSL system (High Linear Power Density Limitation function) is based on the readings of the SPNDs and fixed incore instrumentation.

Twelve fuel assemblies are instrumented with SPND fingers which are distributed radially over the core such that their signals are representative of the key core parameters for different perturbation modes and fuel management schemes. Each of the twelve SPND fingers contains six detectors. In each finger, three SPNDs are located in the top core half and the other three in the bottom core half to detect the peak power density occurring in either the top or bottom halves. Thus they can cover all possible power distributions normal or transient. Axial locations are always situated between two grids to rule out the effect of flux depression in the vicinity of the grids.

Flux mapping is performed periodically with the Aeroball Measurement System (AMS), including reference heat balance, to provide an accurate image of the absolute (i.e. in kW/ft) 3D-power distribution. Based on this flux map, each SPND signal is calibrated to the peak power density within its axial slice. After calibration, all twelve SPNDs within the same axial slice therefore provide the same value, which corresponds to the maximum linear power density value for that axial slice.

The SPND signals are also calibrated to reproduce the power distribution of the hot channel, for minimum DNBR calculation (see LCO 3.2.2 "Departure from Nucleate Boiling Ratio"), and to reproduce the average axial power of each core half, for AXIAL OFFSET calculation (see LCO 3.2.4 "AXIAL OFFSET (AO)").

BASES

BACKGROUND (continued)

The SPND signal gradually increases (conservative) and the gain constants must be periodically recalibrated to prevent unnecessary LPD penalties. Likewise, core burnup can produce power distribution changes that result in non-conservative SPND signals and also require periodic recalibration. The setpoint analyses account for the uncertainty inherent for a given AMS calibration frequency.

The maximum linear power density is monitored continuously by the "High Linear Power Density LCO" function of the RCSL System. Separate LCO setpoints exist for both the upper and lower half of the core. Violation of the linear power density operating limit initiates the following automatic and staggered countermeasures:

First High LPD LCO 1 level

- Audible alarm in the control room
- Prevent dilution signal (only for LPD LCO 1 signal in lower half of the core)
- RCCA bank withdrawal blocking signal
- Turbine generator power increase blocking signal
- RCCA bank insertion blocking signal (only for LPD LCO 1 signal in lower half of the core)

Second High LPD LCO 2 level

- Reduce turbine generator power signal
- Insert RCCA bank signal (only for LPD LCO 2 signal in upper half of the core)

The surveillance setpoint corresponds to the first LPD LCO 1 threshold. The objective of these staggered actions is to prevent operations leading to a further increase of linear power density so that the maximum LPD value can be quickly restored to below its limit.

During power operation with the RCSL System not in service, LPD signals from the PS may be manually monitored to ensure LCO limits are maintained. In this case the automatic and staggered countermeasures described above will not occur and the operator must manually take action to control the LPD.

BASES

APPLICABLE SAFETY ANALYSES	The power distribution and RCCA insertion and alignment LCOs prevent core power distributions from reaching levels that violate acceptance criteria regarding fuel design and coolability. The power density at any point in the core must be limited to maintain the fuel design criteria. This is accomplished by maintaining the power distribution and reactor coolant conditions such that the peak power density and minimum DNBR are within operating limits supported by accident analyses (Ref. 1). Maximum LPD limit assumed in the LOCA analysis (Ref. 1) is typically limiting relative to the maximum LPD assumed in safety analyses for other AOO and postulated accidents. Therefore this LCO provides conservative limits for other AOOs such as uncontrolled RCCA bank withdrawal. Fuel cladding damage does not typically occur while the unit is operating at conditions outside the limits of this LCO during normal operation. Fuel cladding damage could result, however, if an AOO event occurs from initial conditions outside the limits of this LCO. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking during the transient. LPD satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii) (Ref. 3).
LCO	The LOCA safety analysis generally determines the maximum permitted linear power density for the upper half of the core. The LCO limit ensures that the post-LOCA fuel cladding temperature does not exceed a specified maximum limit of 2200°F. As a consequence the LCO ensures that the maximum LPD in the core is not exceeded in the event of a LOCA. A separate limit is also provided for the lower half of the core. This limit provides margin for those events that result in a axial redistribution of power towards the bottom of the core. Both limits ensure margin to fuel centerline melt and maintain clad strain < 1% during all AOOs. The LPD limits are provided in the COLR.
APPLICABILITY	Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 10% RTP. This LCO is not a concern below 10% RTP because the core is operating well below its thermal limits.

BASES

ACTIONS

A.1

With the LPD exceeding its limit, excessive fuel damage could occur following an AOO or postulated accident. In this condition, prompt action must be taken to restore the LPD to within the specified limits.

The one hour limit to restore the LPD to within its specified limits is reasonable since the likelihood of an accident happening over this short period is negligible. The one hour Completion Time also allows the operator sufficient time for evaluating core conditions and confirming automatic actions have been effective or initiating proper corrective actions to restore the LPD to within its specified limits.

B.1

If the value of LPD is not restored to within its limits within the required Completion Time; the unit must be brought in a MODE or condition where the LCO is no longer applicable. This is done by placing the plant in at least MODE 1 with THERMAL POWER < 10% RTP within 6 hours.

The allowed Completion Time of 6 hours is reasonable to reach MODE 1 with THERMAL POWER < 10% RTP from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

The Surveillance requires the operator to verify that the LPD is within limits. This verification is in addition to the automatic checking performed by the RCSL System. The Surveillance can be performed by obtaining the current LPD generated by the RCSL System (providing the RCSL System is in service and has been properly calibrated) and verifying the value is within limits specified in the COLR. Alternately, the verification may also be performed by manually monitoring each OPERABLE PS LPD division and verifying the value is within limits specified in the COLR. Since there are four different divisions based on individual loop conditions, it is necessary to monitor the most limiting LPD division. A 12 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the LPD limits.

REFERENCES

1. FSAR Chapter 15.

2. FSAR Chapter 6.

3. 10 CFR 50.36, Technical Specifications, August, 2007.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid specifically for the loss of cooling accident (LOCA) analyses. An addition this limit allows for further constraining the initial operating conditions assumed in other accident analyses. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the highest integrated linear power along any fuel rod to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the Aeroball Measurement System (AMS). Specifically, the results of a three dimensional power distribution map are analyzed by computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 15 effective full power days (EFPD). However, during power operation, the global power distribution is continuously monitored by LCO 3.2.4, "AXIAL OFFSET (AO)," and LCO 3.2.5, "AZIMUTHAL POWER IMBLANCE (API)," which address directly and continuously measured process variables.

Since DNBR and LPD are monitored independently and protected with separate LCOs which specifically account for the 3D power distribution in the core, $F^N_{\Delta H}$ limits are used to verify the acceptability of the resulting limiting peak cladding temperatures that are used in the LOCA safety analyses.

Operation outside the LCO limits may produce unacceptable consequences if an anticipated operation occurrence (AOO) or other postulated accident occurs.

BASES

APPLICABLE SAFETY ANALYSES	This LCO provides limits on $F^N_{\Delta H}$ for the following purposes: a. Restrict initial LPD to a value which ensures that during a LOCA the peak clad temperatures do not exceed 2200°F; and b. Limit the scope of power distributions from which an accident may be initiated for all FSAR Chapter 15 events. The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 1). $F^N_{\Delta H}$ shall be maintained within the limits specified in the COLR. The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a departure from nucleate boiling (DNB). The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses. $F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii) (Ref. 2).
LCO	$F^N_{\Delta H}$ shall be maintained within the limits specified in the COLR. The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB. The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.
APPLICABILITY	MODE 1 with THERMAL POWER > 90% RTP. Applicability in MODE 1 with THERMAL POWER ≤ 90% RTP is not required because this LCO applies only to LOCA analyses. LOCA events are limiting at HFP because at lower powers there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to challenge licensing criteria.

BASES

ACTIONS

A.1 and A.2

When $F_{\Delta H}^{N}$ exceeds its limit there is little concern regarding DNBR or LPD since these parameters are independently monitored and protected with other trips. However, the LOCA analyses assume this limit at the initiation of the transient therefore exceeding it could result in peak clad temperatures in excess of the acceptance criteria.

The 1 hour limit to reduce power by 1% for each 1% that $F_{\Delta H}^{N}$ limit is exceeded by allows for an orderly power reduction that reduces the hot fuel rod integrated power to near its 100% limit.

The 4 hour limit then provides adequate time to confirm that $F_{\Delta H}^{N}$ has been restored or make necessary adjustments through control rod movements or further power reductions. This completion time also provides a reasonable limit on the amount of time which the plant may outside the $F_{\Delta H}^{N}$ limit.

B.1

When the Required Action cannot be met or completed within the required Completion Time; the unit must be brought in a MODE or condition where the LCO is no longer applicable. This is done by placing the plant in at least MODE 1 with THERMAL POWER ≤ 90% RTP.

The allowed Completion Time of 2 hours is reasonable to reach MODE 1 with THERMAL POWER ≤ 90% RTP from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F_{\Delta H}^{N}$ is determined by taking an AMS flux map. A data reduction computer program (POWERTRAX™) then calculates the maximum value of $F_{\Delta H}^{N}$ from the measured flux distribution. The measured value of $F_{\Delta H}^{N}$ must be multiplied by the appropriate measurement uncertainty before making comparisons to the $F_{\Delta H}^{N}$ limit.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Confirming $F^N_{\Delta H}$ in MODE 1 after an outage and before exceeding 98% power ensures that plant is operating within the limit given the major change in power distributions resulting from the core reload.

The 15 EFPD frequency between $F^N_{\Delta H}$ confirmations is also acceptable since power distributions change relatively slowly over this amount of fuel burnup. Accordingly, this frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after exceeding 90% power. This time period allows sufficient time to perform the required surveillance.

REFERENCES

1. 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, August, 2007.

2. 10 CFR 50.36, Technical Specifications, August, 2007.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 Departure from Nucleate Boiling Ratio (DNBR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss of Coolant Accident (LOCA), loss of flow accident including shaft break, rod ejection accident, or other postulated accident requiring termination by a Protection System (PS) trip function (Ref. 1). This LCO limits the damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., RCCA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting axial power redistribution over time also minimizes the xenon distribution swings, which is a significant factor in controlling axial power distribution.

The power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits for power distribution is accomplished by maintaining the local Linear Power Density (LPD) and the DNBR within limits.

Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux.

There are two systems that perform online monitoring of the core maximum LPD and minimum DNBR:

- The Protection System (PS) and;
- The Reactor Control, Surveillance and Limitation (RCSL) system.

BASES

BACKGROUND (continued)

The PS and the RCSL system are capable of verifying that the LPD and the DNBR do not exceed their limits. The PS and the RCSL system perform this function by continuously monitoring incore Self-Powered Neutron Detectors (SPND) measurements, thermal-hydraulic data, and Rod Cluster Control Assemblies (RCCA) insertion, and by calculating an actual value of DNBR and LPD, for comparison to:

- The respective trip setpoints in the PS and;
- The limits of acceptable operation in the RCSL system.

Thus the RCSL system indicates continuously to the operator how far the core is from the operating LPD and DNBR limits, and provides an alarm in the Main Control Room if any limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an AOO or postulated accident, but does not necessarily imply a violation of fuel design limits.

The calculation of the minimum DNBR in the PS (Low DNBR trip function) and in the RCSL system (Low DNBR LCO function) is based for both systems on:

- The power density distribution in the hot channel, which is based on the readings of the SPND (reconstruction) fixed incore instrumentation;
- The inlet reactor coolant temperature;
- The pressurizer pressure and;
- The RCS flow rate.

Twelve fuel assemblies are instrumented with SPND fingers which are distributed radially over the core such that their signals are representative of the key core parameters for different perturbation modes and fuel management schemes. Each of the twelve SPND fingers contains six detectors. In each finger, three SPNDs are located in the top core half and the other three in the bottom core half to detect the peak power density occurring in either the top or bottom halves. Thus they can cover all possible power distributions normal or transient. Axial SPND locations are always situated between two grids to rule out the effect of flux depression in the vicinity of the grids.

Flux mapping is performed periodically with the Aeroball Measurement System (AMS), including reference heat balance, to provide an accurate image of the absolute (i.e. in kW/ft) 3D-power distribution. In each finger, the six SPND signals are then calibrated to the power density of the hot

BASES

BACKGROUND (continued)

rod integrated on the length of the SPND. After calibration, all twelve SPND fingers therefore provide the same axial power shape representative of the power shape of the actual hot channel.

The SPND signal gradually increases (conservative) with core burnup and the gain constants must be periodically recalibrated to prevent unnecessary DNBR penalties. Likewise, core burnup can produce power distribution changes that result in non-conservative SPND signals and also require periodic recalibration. Setpoint analyses account for the uncertainty inherent for a given AMS calibration frequency.

The PS and the RCSL system use these measurements and a proprietary algorithm to reconstruct the local thermal-hydraulic conditions at the minimum DNBR point in the core and apply the chosen Critical Flux Predictor (Ref. 3) to calculate the DNBR.

The minimum DNBR is monitored continuously by the "Low DNBR LCO" function of the RCSL system and indicated to the operator. Violation of the DNBR operating limit initiates the following automatic and staggered countermeasures:

First Low DNBR LCO 1 level
- RCCA bank withdrawal blocking signal, and
- Turbine generator power increase blocking signal

Second Low DNBR LCO 2 level
- Reduce turbine generator power signal, and
- Insert RCCA bank signal

The surveillance setpoint corresponds to the first Low DNBR LCO threshold. The objective of these staggered actions is to prevent operations leading to a further decrease of the DNBR such that the minimum DNBR value can be quickly restored to above its limit.

During power operation with the RCSL system out of service, DNBR signals from the PS may be manually monitored to ensure LCO limits are maintained. In this case the automatic and staggered countermeasures described above will not occur and the operator must manually take action to control the LPD. In addition, since each the PS division signals are derived using only one loops signals (rather than averaging them as is done in RCSL), the PS signal has a higher measurement uncertainty. This must be accounted for by monitoring to a higher DNB LCO limit as specified in the COLR.

BASES

APPLICABLE SAFETY ANALYSES

The power distribution and RCCA insertion and alignment LCOs prevent core power distributions from reaching levels that violate acceptance criteria regarding fuel design and coolability. The DNBR at any point in the core must be limited to maintain the fuel design criteria. This is accomplished by maintaining the power distribution and reactor coolant conditions such that the peak power density and minimum DNBR are within operating limits supported by accident analyses.

The minimum DNBR limit is typically established based on the Loss of Coolant Flow accident which is limiting relative to the maximum ΔDNBR assumed in safety analyses for all other AOOs. Therefore this LCO provides conservative limits for all other AOOs.

Fuel cladding damage does not normally occur while the unit is operating at conditions outside the limits of this LCO during normal operation. Fuel cladding damage could result, however, if an AOO event occurs from initial conditions outside the limits of this LCO. The potential for fuel cladding damage exists because changes in the power distribution can cause a reduction in DNB margin at the initiation of a fast transient such that other plant trips can no longer respond in time to protect the fuel design limits.

DNBR satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii) (Ref. 4).

LCO

The Loss of Flow accident generally establishes the DNB LCO limits as this transient is too fast to be protected by the Low DNBR trip in the PS. The DNB LCO therefore ensures that the plant operates far enough away from the DNBR design limit that in the event of a very fast transient sufficient time exists for other plants trips to intervene prior to exceeding the DNBR design limit. The DNBR limits are provided in the COLR.

APPLICABILITY

The DNB LCO is only applicable in MODE 1 above 10% RTP. This LCO is not a concern below 10% RTP and for lower operating MODES because the stored energy in the fuel and the energy being transferred to the reactor coolant are sufficiently low that DNBR is no longer a concern.

ACTIONS

A.1

With the DNBR exceeding its limit, excessive fuel damage could occur following an AOO or postulated accident. In this condition, prompt action must be taken to restore the DNBR to within the specified limits.

BASES

ACTIONS (continued)

The 1 hour limit to restore the DNBR to within its specified limits is reasonable since the likelihood of an accident happening over this short period is negligible. The 1 hour Completion Time also allows the operator sufficient time for evaluating core conditions and either confirm automatic actions have been effective or initiating proper corrective actions to restore the DNBR to within its specified limits.

B.1

If the value of DNBR is not restored within its specified limits within the required Completion Time; the unit must be brought in a MODE or condition where the LCO is no longer applicable. This is done by placing the plant in at least MODE 1 with THERMAL POWER ≤ 10% RTP within 6 hours.

The allowed Completion Time of 6 hours is reasonable to reach MODE 1 with THERMAL POWER ≤ 10% RTP from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The Surveillance requires the operator to verify that the DNBR is within limits. This verification is in addition to the automatic checking performed by the RCSL System. The Surveillance can be performed by obtaining the current DNBR generated by the RCSL System (providing the RCSL System is in service and has been properly calibrated) and verifying the value is within limits specified in the COLR. Alternately, the verification may also be performed by manually monitoring each OPERABLE PS DNBR division and verifying the value is within limits specified in the COLR. Since there are four different divisions based on individual loop conditions, it is necessary to monitor the most limiting DNBR division. A 12 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the DNBR limits.

BASES

REFERENCES

1. FSAR Chapter 15.
2. FSAR Chapter 6.
3. ANP-10269P and Supplement 1, Rev. 0, August, 2007.
4. 10 CFR 50.36, Technical Specifications, August, 2007.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 AXIAL OFFSET (AO)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected rod cluster control assembly (RCCA) accident, or other postulated accident requiring termination by a Protection System (PS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient (Ref. 2).

AXIAL OFFSET (AO) is a measure of the axial power distribution in the core. The purpose for a limit on AO is to limit the axial power distributions to initial values assumed in the accident analyses. Extreme shifts in power towards either the top or bottom of the core can have adverse impacts during an accident. In general, top-peaked power shapes have lower minimum departure from nucleate boiling ratio(s) (MDNBRs) to start with while bottom-peaked shapes tend to result in more significant DNBR degradation during a transient. Significant shifts in either direction can lead to increased linear power densities (LPDs). Minimizing power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

The reactor control, surveillance and limitations (RCSL) system continuously monitors the AXIAL OFFSET based on evaluations of the core power distribution using the incore self-powered nuclear detectors (SPNDs). Twelve fuel assemblies are instrumented with SPND fingers which are distributed radially over the core such that their signals are representative of the key core parameters for different perturbation modes and fuel management schemes. Each of the twelve SPND fingers contains six detectors. In each finger, three SPNDs are located in the top core half and the other three in the bottom core half to detect power density occurring in top and bottom halves. Thus they can cover all possible power distributions, normal or accidental. Axial locations are always between two grids to rule out the effect of flux depression in the vicinity of the grids.
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BASES

BACKGROUND (continued)

The SPND signal gradually increases (conservative) with core burnup and the gain constants must be periodically recalibrated to prevent unnecessary AO penalties. Likewise, core burnup can produce power distribution changes that result in non-conservative SPND signals and also require periodic recalibration. Setpoint analyses account for the uncertainty inherent for a given AMS calibration frequency.

The AXIAL OFFSET is monitored continuously by the "Axial Power Shape LCO" function of the RCSL system. The Axial Power Shape LCO function aims at informing the operator if the AO limit is violated. The AO setpoint is function of the core thermal power level. Violation of the AO operating limits initiates an alarm in the main control room and an automatic signal blocking any turbine generator power increase. Active countermeasures are not initiated so as not to interfere with those automatic actions initiated by the "AXIAL OFFSET Control" function in RCSL, which will tend to restore the AO to within limits.During power operation with the RCSL system out of service, SPND signals from the PS may be manually monitored to determine AO and verify the LCO limit is maintained.

APPLICABLE SAFETY ANALYSES

The maximum AO limit is established for the following purposes:

a. Restrict initial AO to a value which ensures that during a LOCA the peak clad temperatures do not exceed 2200°F (Ref. 1) and;

b. Restrict the scope of power distributions assumed as initial conditions in analyzing anticipated operational occurrences (AOOs) and postulated accidents.

Fuel cladding damage does not typically occur while the unit is operating at conditions outside the limits of this LCO during normal operation. Fuel cladding damage could result, however, if an AOO event occurs from initial conditions outside the limits of this LCO. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking during the transient.

AO satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii) (Ref. 3).

BASES

LCO

The positive AO limit is generally established to minimize or eliminate the consequences of the rod ejection or uncontrolled RCCA withdrawal transient. The negative AO limit is generally established by The Loss of Flow transient. The limits are established around a target AO that is a function of the core management scheme. The AO limits are provided in the COLR.

Violation of this LCO could produce unacceptable consequences if an AOO or postulated accident occurs while the AO is outside its specified limits.

APPLICABILITY

The AO LCO is only applicable in MODE 1 above 50% RTP. This LCO is not a concern below 50% RTP and for lower operating MODES because xenon transients generated within the lower power level range are not severe. In addition, significant margin to thermal limits exists at lower power levels and therefore thermal limits are not significantly challenged.

ACTIONS

A.1

With the AO exceeding its limit, excessive fuel damage could occur following an AOO or postulated accident. In this condition, prompt action must be taken to restore the AO to within the specified limit.
The 1 hour time period to restore the AO to within its specified limit is reasonable and ensures that the core does not continue to operate in this Condition. The 1 hour Completion Time also allows the operator sufficient time for evaluating core conditions and confirming automatic actions have been effective or initiating proper corrective actions to restore the AO to within its specified limit.

B.1

If the value of AO is not restored to within its specified limit within the required Completion Time; the unit must be placed in a MODE or condition where the LCO is no longer applicable. This is done by placing the plant in a least MODE 1 with THERMAL POWER to < 50% RTP within 4 hours.

The allowed Completion Time of 4 hours is reasonable to reach MODE 1 with THERMAL POWER < 50% RTP from full power operation in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

The Surveillance requires the operator to verify that the AO is within limits. This verification is in addition to the automatic checking performed by the RCSL System. The Surveillance can be performed by obtaining the current AO generated by the RCSL System (providing the RCSL System is in service and has been properly calibrated) and verifying the value is within limits specified in the COLR. Alternately, the verification may also be performed by manually monitoring each OPERABLE PS AO division and verifying the value is within limits specified in the COLR. Since there are four different divisions based on individual loop conditions, it is necessary to monitor the most limiting AO division. A 12 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the AO limits.

Another option is to monitor the AO through the generation of an AMS flux map. A data reduction computer program (POWERTRAX™) then calculates the core wide assembly nodal power distribution from the measured flux distribution.

REFERENCES

1. 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, August, 2007.

2. FSAR Chapter 15.

3. 10 CFR 50.36, Technical Specifications, August, 2007.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 AZIMUTHAL POWER IMBALANCE (API)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to those assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss of Coolant Accident (LOCA), loss of flow accident, rod ejection accident, or other postulated accident requiring termination by a Protection System (PS) trip function (Ref. 1). This LCO limits damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

API > 1.04 are not expected during normal operation. If they do occur then actions must be taken since calculation of many core parameters (e.g. rod worths, moderator temperature coefficient (MTC), etc) do not explicitly account for azimuthal power asymmetries. Core API that are too large could invalidate the uncertainties assumed for these parameters. In addition, a large API indicates the existence of potential adverse phenomena in the core which warrant further evaluation.

APPLICABLE SAFETY ANALYSES

This LCO precludes power distributions that violate the following design criteria:

- The post-LOCA fuel cladding temperature does not exceed a specified maximum limit of 2200°F.
- There will be at least a 95% probability with 95% confidence that the hot rod in the core will not experience a DNB condition.
- The energy deposition to the fuel during a rod ejection accident will not exceed the limits specified in Reference 3.
- Control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod fully withdrawn.

The LCO limits on the AO, API, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions from occurring which would exceed the safety analyses limits or invalidate assumptions used in deriving core analytical parameters. Fuel cladding damage could result if an AOO event occurs from initial conditions outside the limits of this LCO. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking during the transient as well as changes in parameters such as rod worths which affect SDM.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The API in the core must be limited to maintain the fuel design criteria. This is accomplished by maintaining the power distribution and reactor coolant conditions such that the peak power density and API are within operating limits supported by accident analyses.

API satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii) (Ref. 3).

LCO

Since the PS and RCSL sense power distributions from the incore SPNDs, they both are able to account for the effects of core radial power asymmetries. The PS automatically provides for reductions in the Low Departure from Nucleate Boiling (DNBR) trip, whenever a significant API is detected. Most analytical parameters generated for safety analyses though are determined assuming symmetric core radial power distributions. Asymmetries in these distributions can impact parameters such as stuck rod worths and SDM. An API limit of ≤ 1.04 is sufficient to ensure the validity of these assumptions and therefore the licensing analyses.

APPLICABILITY

The API LCO is only applicable in MODE 1 above 50% RTP. This LCO is not a concern below 50% RTP and for lower operating MODES because the effect of radial power asymmetries is reduced as power levels are reduced and therefore become insignificant at these lower power levels. In addition, the stored energy in the fuel and the energy being transferred to the reactor coolant are low at these lower power levels.

ACTIONS

A.1

With API exceeding its limit, action must be taken to restore API within its limit within 2 hours. Restoring the API to within its specified limit of ≤ 1.04 in 2 hours is reasonable since the likelihood of an accident happening over this short period is negligible and it ensures that the core does not continue to operate in this condition. The 2 hour Completion Time also allows the operator sufficient time for evaluating core conditions and determining the cause of the API problem.

BASES

ACTIONS (continued)

B.1

If API is not restored to within its limit within the required Completion Time; the unit must be placed in a MODE or condition where the LCO is no longer applicable. This is done by placing the plant in a least MODE 1 with THERMAL POWER to < 50% RTP within 4 hours.

The allowed Completion Time of 4 hours is reasonable to reach MODE 1 with THERMAL POWER < 50% RTP from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

The Surveillance requires the operator to verify that the API is within limits. This verification is in addition to the automatic checking performed by the RCSL System. The Surveillance can be performed by obtaining the current API generated by the RCSL System (providing the RCSL System is in service and has been properly calibrated) and verifying the value is within limits specified in the COLR. Alternately, the verification may also be performed by manually monitoring each OPERABLE PS API division and verifying the value is within limits specified in the COLR. Since there are four different divisions based on individual loop conditions, it is necessary to monitor the most limiting API division. A 12 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the API limits.

Another option is to monitor the API through the generation of an AMS flux map. A data reduction computer program (POWERTRAX^{TM}) then calculates the core wide assembly nodal power distribution from the measured flux distribution.

REFERENCES

1. FSAR Chapter 15.

2. 10 CFR 50.36, Technical Specifications, August, 2007.

B 3.3 INSTRUMENTATION

B 3.3.1 Protection System (PS)

BASES

BACKGROUND

The PS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary during anticipated operational occurrences (AOOs). The PS also initiates the Engineered Safety Features (ESF) actuations that are used to mitigating accidents. The ESF actuates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits, maintain the Reactor Coolant System (RCS) pressure boundary, and mitigate the consequences of accidents that could result in potential exposures comparable to the guidelines set forth in 10 CFR 100 during AOOs and ensures acceptable consequences during accidents.

The PS initiates and the Safety Automation System (SAS) controls the necessary safety systems to protect against violating core design limits, maintain the RCS pressure boundary, and mitigate the consequences of accidents that could result in potential exposures comparable to the guidelines set forth in 10 CFR 100 during anticipated operational occurrences and ensures acceptable consequences during postulated accidents.

The four redundant divisions of the PS are physically separated in their respective safeguard buildings. The four divisionally separated rooms containing the PS equipment are in different fire zones. Therefore, in general, the consequences of internal hazards (e.g., fire), would impact only one PS division.

The PS architecture is four-fold redundant for both reactor trip and ESF functions. A single failure during corrective or periodic maintenance, or a single failure and the effects of an internal hazard does not prevent performance of the safety functions. For the reactor trip functions, each PS division actuates one division of the reactor trip devices based on redundant processing performed in four divisions. For ESF functions, the redundancy of the safety function as a whole is defined by the redundancy of the ESF system mechanical trains. In general, this results in one PS division actuating one mechanical train of an ESF system based on redundant processing performed in four divisions. The PS not only supports the redundancy of the mechanical trains, but also enhances this redundancy through techniques such as redundant actuation voting.

BASES

BACKGROUND (continued)

Three of the four divisions are necessary to meet the redundancy and testability of GDC 21 in 10 CFR 50, Appendix A (Ref. 3). The fourth division provides additional flexibility by allowing one division to be removed from service for maintenance or testing while still maintaining a minimum two-out-of-three logic. Thus, even with a division inoperable, no single additional failure in the PS can either cause an inadvertent trip/ESF or prevent a required trip/ESF from occurring.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the PS, as well as LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to contain LSSS defined by the regulation as "...settings for automatic protective devices...so chosen that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protective devices must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur. When LSSS is specified for a variable having a significant safety function, but which does not protect SLs, the setting must be chosen such that automatic protective actions will initiate consistent with the design basis. The Design Limit is the limit of the process variable at which the safety action is initiated to ensure that these automatic protective devices will perform their specified safety function. These limits (i.e., the Analytical Limits and Design Limits) constitute the Setting Basis specified in Table 3.3.1-2.

BACKGROUND (continued)

The LTSP is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable reaching the Analytical/Design Limit and thus ensuring that the SL would not be exceeded (i.e., for Analytical Limits) or that automatic protective actions occur consistent with the design basis (i.e., for Design Limits). As such, the LTSP accounts for uncertainties in setting the device (e.g., CALIBRATION), uncertainties in how the device might actually perform (e.g., repeatability), changes in the point of action of the device over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the LTSP ensures that SLs are not exceeded and that automatic protective devices will perform their specified safety function.

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety function(s)." For automatic protective devices, the required safety function is to ensure that the SL is not exceeded and that automatic protective actions will initiate consistent with design basis. However, use of the LTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protective device setting during a Surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protective device with a setting that has been found to be different from the LTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the LTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded or that automatic protective actions would initiate consistent with the design basis with the "as-found" setting of the protective device. Therefore, the device would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the device to the LTSP to account for further drift during the next surveillance interval.

BASES

BACKGROUND (continued)

However, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. This value is specified in the SCP, as required by Specification 5.5.18, in order to define OPERABILITY of the devices and is designated as the Allowable Value, which is the least conservative value of the as-found setpoint that a division can have during a periodic CALIBRATION or SENSOR OPERATIONAL TEST.

The actual LTSP and Allowable Values (derived for the Setting Basis values specified in Table 3.3.1-2) and the methodology for calculating the as-found and as-left tolerances are maintained in SCP, as required by Specification 5.5.18.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling (DNB),

- Fuel centerline melting shall not occur; and

- The RCS pressure SL of 2803 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within 10 CFR 100 limits. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The PS is segmented into four interconnected modules and associated LCOs for the reactor trips and ESF functions. These modules are:

- Sensors, which include the associated instrumentation;
- Manual actuation switches;

BASES

BACKGROUND (continued)

- Signal Processors, which include:
  - Remote Acquisition Units (RAUs), which acquire the signals from the Self-Powered Neutron Detectors (SPND) and distribute these signals;
  - Acquisition and Processing Units (APUs), which perform calculations and make setpoint comparisons; and
  - Actuation Logic Units (ALUs), which perform voting of the processing results from the redundant APUs in the different divisions and to issue actuation orders based on the voting results; and
- Actuation Devices, which includes the reactor trip breakers and contactors and the Priority Actuation and Control Systems (PACS) control modules for the Reactor Coolant Pump (RCP) bus and trip breakers..

The PS is a digital, integrated reactor protection system and engineered safety features actuation system. Individual sensors, signal processors, or the ALUs that provide the actuation signal voting function, can be associated with multiple reactor trip, ESF functions, and Permissives.

<u>Sensors</u>

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

The Power Density Detector System, which uses SPND and RAUs, provides the in-core monitoring function. The Power Range, Intermediate Range, and Source Range monitors provide the ex-core monitoring functions.

The instrument setpoint methodologies are discussed in the SCP.

BASES

BACKGROUND (continued)

The SCP ensures that appropriate settings are used for Trip/Actuation Functions and that SLs of Chapter 2.0, "Safety Limits (SLs)," are not violated during AOOs, and the consequences of postulated accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or postulated accident and the equipment functions as designed.

Functional testing of the entire PS, from sensor input through the opening of individual sets of Reactor Trip Circuit Breakers (RTCB) or contactors, is performed each refueling cycle. Processing transmitter CALIBRATION is also normally performed on a refueling basis.

Manual Actuation Switches

Manual controls necessary to perform the manual operator actions credited in the safety analysis are included within the scope of the Technical Specifications. Manual actuation switches are provided to initiate the reactor trip function from the main control room (MCR) and the remote shutdown station (RSS). The ability to manually initiate ESF systems is provided in the MCR. Manual actuation of ESF systems initiates all actions performed by the corresponding automatic actuation including starting auxiliary or supporting systems and performing required sequencing functions.

Signal Processors

The PS is a distributed, redundant computer system. It consists of four independent redundant data-processing automatic paths (divisions), each with layers of operation and running asynchronous with respect to each other. In addition to the computers associated with the automatic paths, there are two redundant message and service interface computers to interface with each division.

The measurement channels or signal acquisition layer (which includes the RAUs) in each division acquires analog and binary input signals from sensors in the plant (such as for temperature, pressure, and level measurements). Each signal acquisition computer distributes its acquired and preprocessed input signals to the PS logic and controls, which includes the data processing computers (APUs).

BASES

BACKGROUND (continued)

The data-processing computers (APUs) perform signal processing for plant protective functions such as signal online validation, limit value monitoring and closed-loop control calculations. Each PS division contains four ALUs, two assigned to each subsystem. Two ALUs of the same subsystem within a division are redundant and perform the same processing using the same inputs. The outputs of two redundant ALUs are combined in a hardwired “functional AND” logic for reactor trip functions and in a hardwired OR logic for ESF functions. This avoids both unavailability of ESF functions and spurious reactor trips. The data processing computers then send their outputs to two independent voter computer units (ALUs) in each division.

In the voter computers, the outputs of the data-processing computers of redundant (three or four) divisions are processed together. A voter computer controls a set of actuators. Each voter receives the actuation signal from each of the redundant data-processing computers. The voter's task is to compare this redundant information and compute a validated (voted) actuating signal, which is used for actuating the end devices.

Each PS division contains four ALUs, two assigned to each subsystem. The two ALUs of the same subsystem within a division are redundant and perform the same processing using the same inputs. The outputs of two redundant ALUs are combined in a hardwired “functional AND” logic for reactor trip functions and in a hardwired OR logic for ESF functions.

For the reactor trip function, both ALUs in a division, if OPERABLE, must vote for an actuation. This provides protection against spurious trips. However, if only one ALU in a division is OPERABLE, the division is still OPERABLE, and the single voting ALU will initiate a reactor trip. For the ESF functions, an actuation will occur if either of the ALUs in a division votes for an actuation. This provides protection against ESF unavailability.

Reactor Trip Logic

Critical plant parameters such as temperatures, pressures, and levels are sensed, acquired, and converted to electrical signals by the PS. These signals are sent to various reactor trip functions in the PS where they are processed. When prohibited operating conditions exist, a reactor trip signal is generated from the reactor trip functions. Besides being generated automatically from the PS, a reactor trip signal can also be generated from the following systems:

BASES

BACKGROUND (continued)

- Automatic reactor trip from SAS in the event that the PS is lost;
- Manual trip from the Safety Information and Control System (SICS) panel. Four reactor trip switches are provided, which correspond to each of the four divisions;
- Manual trip from the control room; and
- Manual trip from the RSS. Note that the RSS manual trip is not part of the required circuits for LCO 3.3.1.

The reactor trip functions will utilize voting logic in order to screen out potential upstream failures of sensors or processing units. The architecture of the PS, as well as logic implemented in the system, will guard against spurious reactor trip orders while ensuring that those orders will be available when needed.

Single failures upstream of the ALU layer that could result in an invalid signal being used in the reactor trip actuation are marked as faulted by modifying the vote in the ALU layer. For the reactor trip functions, the vote is always modified toward actuation.

ESF Trip Logic

The ESF trip logic senses accident situations and initiate the operation of necessary features. The ESF along with reactor trip ensure the following:

- The integrity of the reactor coolant pressure boundary;
- The capability to shut down the reactor and maintain it in a safe shutdown condition; and
- The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures.

BASES

BACKGROUND (continued)

As with the reactor trip logic, critical plant parameters such as temperatures, pressures, and levels are sensed, acquired, and converted to electrical signals by the PS. When prohibited operating conditions exist, an ESF signal is generated from the PS. In addition to the automatic ESF actuation functions performed by the PS, the capability to manually initiate these functions is provided in the MCR. These manual functions are implemented at the system level and perform the same actions as the automatic functions. The implementation of manual system level actuation of ESF functions and the priority between the automatic functions of the PS and the manual system level initiation is determined on a case-by-case basis.

Single failures upstream of the ALU layer that could result in an invalid signal being used in the ESF actuation are marked as faulted by modifying the vote in the ALU layer. For the ESF functions, the vote is modified toward actuation except:

- The Main Steam Relief Train (MSRT) divisions, which degrade towards isolation; and
- Pressurizer Safety Relief Valve (PSRV) opening for cold overpressure protection, which degrades towards non-actuation.

<u>Actuation Devices</u>

Reactor Trip Actuation Devices

The reactor trip actuation is performed by interrupting electrical power to the Control Rod Drive Mechanisms (CRMD). Electrical power to the CRDM is delivered by the Control Rod Drive Power Supply System (CRDPSS). The CRDPSS consists of 220 V DC distribution boards which are fed from the Uninterruptible Power Supply System.

BASES

BACKGROUND (continued)

The power supply of the CDRM can be switched off via the following features:

- Four main trip breakers distributed in two electrical divisions. Two breakers are located in Division 2, two others in Division 3. The main trip breakers can be opened by two coils: one with a de-energized logic using an under voltage coil and the other with an energized logic using a shunt trip coil.

- Four trip contactors combined in a 2-out-of-4 logic feed a group of four CRDM. Division 1, 2, and 3 contains eleven groups of four CRDMs. Division 4 contains eleven groups of four CRDMs and one single CRDM for the central rod. There are a total of 92 contactors. Each trip contactor is switched off by a de-energized coil.

- The electronics of the RodPilot can switch-off the power supply of four CRDMs. Two groups of four commands can actuate this electronic module, one with low active and one with high active logic. The electronics of the RodPilot is a non-safety device of the reactor trip but is the fastest switching device and allows the contactors and the trip breaker to open without stress.

- The under voltage coil of the main trip breakers is actuated by the automatic reactor trip signals of the PS and the manual trip from the SICS panel. The shunt coil of the main trip breakers is actuated by the automatic reactor trip signal from the SAS and the manual trip signal from the RSS. The shunt coil of the trip breakers receives two different signals from SAS and RSS combined in an "OR" logic performed at the level of trip breakers.

The operator can manually close the breakers by individual controls. This control actuates the closing coil of the breaker via the SAS. In the electronics of the breaker, the opening of trip breaker must have priority to the closing.

The reactor trip signal generated automatically by the PS and the manual trip signal generated from the SICS panel can actuate the trip contactors.

Engineered Safety Features Actuation Devices

The ESF determines the need for actuation in each of the input divisions monitoring each actuation parameter. Once the need for actuation is determined, the condition is transmitted to automatic actuation output logic divisions, which perform the logic to determine the actuation of each end device. Each end device has its own automatic actuation logic.

BASES

BACKGROUND (continued)

Each of the PS sensors, signal processors, or actuation devices can be placed in lockout, which renders the component inoperable. The digital signals within the PS carry a value and a status. The signal status can be propagated through the software function blocks; therefore, if an input signal to a function block has a faulty status, the output of the function block also has a faulty status. When a signal with a faulty status reaches the voting function block, the signal is disregarded through modification of the voting logic. Individual function computers can be put into a testing and diagnostic mode via the service unit. The function processor that is being tested then behaves like a computer with a "detected fault" for the system. The signal outputs are disabled and those sent via the communication means are marked with the status "TEST" or "ERROR" and therefore masked by selection blocks with active status processing. In this case the receiving function processor behaves as if the transmitting function processor had failed.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The PS is designed to ensure that the following operational criteria are met:

– The associated actuation will occur when the parameter monitored by each division reaches its setpoint and the specific coincidence logic is satisfied; and

– Separation and redundancy are maintained to permit a division to be out of service for testing or maintenance while still maintaining redundancy within the PS instrumentation network.

Each of the analyzed transients and accidents can be detected by one or more PS Functions. Each of the PS reactor trip and ESF Functions included in the Technical Specifications are credited as part of the primary success path in the accident analysis. Non-credited functions are purely equipment protective, and their use minimizes the potential for equipment damage. Non-credited functions are not included in the Technical Specifications. Refer to FSAR Sections 7.2 and 7.3.

The LCO requires the PS sensors, manual actuation switches, signal processors, and specified actuation devices to be OPERABLE. The LCO ensures that each of the following requirements is met:

– A reactor trip or ESF function will be initiated when necessary; and

– Sufficient redundancy is maintained to permit a component to be out of service for testing or maintenance.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Failure of any sensors, signal processors, or actuation device reduces redundancy or renders the affected division(s) inoperable.

The Limiting Trip Setpoints, Allowable Values, and as-left and as-found tolerances, and the methodologies to calculate these values are specified in the SCP (Specification 5.5.18).

The PS sensors, manual actuation switches, signal processors, and specified actuation devices satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii) .

The PS sensors, manual actuation switches, signal processors, and specified actuation devices that support reactor trips are required to be OPERABLE in MODES 1, 2 and/or 3 because the reactor is or can be made critical in these MODES. The automatic reactor trip functions are designed to take the reactor subcritical, which maintains the SLs during AOOs and assists the ESF in providing acceptable consequences during accidents. The PS sensors, manual actuation switches, signal processors, and specified actuation devices that support automatic reactor trip functions are not required to be OPERABLE in MODES 4 and 5. In MODES 4 and 5, the emphasis is placed on return to power events. The reactor is protected in these MODES by ensuring adequate SDM.

The PS sensors, manual actuation switches, signal processors, and specified actuation devices that support reactor trips are required to be OPERABLE in MODES 1, 2, 3 and/or 4 since there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the MSIVs to preclude a positive reactivity addition,
- Actuate Emergency Feedwater (EFW) to preclude the loss of the SGs as a heat sink (in the event the normal feedwater system is not available),
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB), and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

In MODES 5 and 6, automatic actuation of the ESF Functions is not normally required because adequate time is available to evaluate plant conditions and respond by manually operating the ESF components if required. Exceptions to this are:

- ESF 10.a - Emergency Diesel Generator (EDG) Start on Degraded Grid Voltage,
- ESF 10.b - EDG Start on Loss of Offsite Power (LOOP),
- ESF 11.b - Chemical and Volume Control System (CVCS) Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating),
- ESF 11.c - CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions,
- ESF 12.a and 12.b - PSRV Actuation - First and Second Valve, and
- ESF 13 - Control Room Heating, Ventilation and Air Conditioning (HVAC) Reconfiguration to Recirculation Mode on High Intake Activity.

These ESF functions are required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies to ensure that:

- Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;
- Systems needed to mitigate a fuel handling accident are available; and
- Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The specific safety analysis and OPERABILITY requirements applicable to each PS protective function is identified below.

A. REACTOR TRIPS

1. Low DNBR (Includes High Outlet Quality)

This function protects the fuel against the risk of departure from nucleate boiling during AOOs that lead to a decrease of the DNBR value. There are five Low DNBR trips:

a. Low DNBR,
b. Low DNBR and Imbalance or Rod Drop,
c. Variable Low DNBR and Rod Drop,
d. Low DNBR - High Quality, and
e. Low DNBR - High Quality and Imbalance or Rod Drop.

Together, these five trips protect against the following AOOs:

- Increase in heat removal by the secondary system,
- Decrease in heat removal by the secondary system,
- Reactivity and power distribution anomalies, and
- Decrease in reactor coolant inventory.

The Low DNBR (1.a) and High Quality (1.d) trips require four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- SPNDs,
- RCP speed sensor,
- Pressurizer Pressure (Narrow Range) sensor,
- Cold leg temperature (Narrow Range) sensor,
- RCS loop flow sensors,
- RAU,
- APUs, and
- ALUs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low DNBR and Imbalance or Rod Drop (1.b), Variable Low DNBR and Rod Drop (1.c), and High Quality and Imbalance or Rod Drop (1.e) trips require four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- SPNDs,
- Rod Cluster Control Assembly (RCCA) position indicators,
- RCP speed sensor,
- Pressurizer Pressure (Narrow Range) sensor,
- Cold leg temperature (Narrow Range) sensor,
- RCS loop flow sensors,
- RAU,
- RCCA Unit,
- APU, and
- ALUs.

The Analytical Limits are high enough to provide an operating envelope that prevents an unnecessary low DNBR reactor trip. The Analytical Limits are low enough for the system to maintain a margin to unacceptable fuel cladding damage for AOOs that leads to an uncontrolled decrease of the DNBR value.

The P2 permissive automatically enables the five Low DNBR Trip signals when the neutron flux, as measured by the power range, is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trips are also automatically disabled by Permissive P2.

2. High Linear Power Density

This function protects the fuel against the risk of melting at the center of the fuel pellet, during accidental transients, for events leading to an uncontrolled increase of the linear power density.

This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Reactivity and power distribution anomalies.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The High Linear Power Density Trip requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- SPNDs,
- RAU,
- APUs, and
- ALUs.

The Analytical Limits are high enough to provide an operating envelope that prevents unnecessary High Linear Power reactor trips. The Analytical Limits are low enough for the system to maintain a margin to unacceptable fuel centerline melt for any AOOs that lead to an uncontrolled increase of the linear power density.

The P2 permissive automatically enables the Reactor Trip signal when the neutron flux, as measured by the power range, is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trip is also automatically disabled by Permissive P2.

3. High Neutron Flux Rate of Change (Power Range)

This function limits the consequences of an excessive reactivity increase from an intermediate power level including nominal power. This trip protects against reactivity and power distribution anomalies.

The High Neutron Flux Rate of Change Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3 with the Reactor Control, Surveillance and Limitation (RCSL) System capable of withdrawing a RCCA or one or more RCCAs not fully inserted:

- Power Range sensors,
- APUs, and
- ALUs.

The Analytical Limit is high enough to provide an operating envelope that prevents unnecessary Excore High Neutron Flux Rate of Change reactor trips. The Analytical Limit is low enough for the system to maintain a margin to unacceptable fuel cladding damage due to an excessive reactivity increase from an intermediate power level including nominal power.

There are no permissives associated with this trip.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. High Core Power Level

This function limits the consequences of an excessive reactivity increase from an intermediate high power level including nominal power. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Reactivity and power distribution anomalies.

The High Core Power Level Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and in MODE 2 when the nuclear power level is greater than or equal to 10^{-5}% power as indicated on the Intermediate Range monitors:

- Cold Leg Temperature sensors (Wide Range),
- Hot Leg Temperature (Narrow Range) sensors,
- Hot Leg Pressure (Wide Range) sensors,
- RCS Loop Flow sensors,
- APUs, and
- ALUs.

The Analytical Limit is high enough to provide an operating envelope that prevents an unnecessary High Core Power Level reactor trip. The Analytical Limit is low enough for the system to maintain a margin to unacceptable fuel cladding damage due to an excessive reactivity increase from an intermediate high power level including nominal power.

The P5 permissive automatically enables the High Core Power Level Trip when the nuclear power level is greater than or equal to 10^{-5}% power. The P5 permissive also automatically disables the High Core Power Level Trip below this power.

5. Low Saturation Margin

This function limits the consequences of an excessive reactivity increase from an intermediate high power level including nominal power. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Reactivity and power distribution anomalies.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low Saturation Margin Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODE 2 when the nuclear power level is greater than or equal to 10^{-5}% power as indicated on the Intermediate Range monitors.:

- Cold Leg Temperature sensors (Wide Range),
- Hot Leg Temperature (Narrow Range) sensors,
- Hot Leg Pressure (Wide Range) sensors,
- RCS Loop Flow sensors,
- APUs, and
- ALUs.

The Design Limit is high enough to provide an operating envelope that prevents an unnecessary Low Saturation Margin reactor trip. The Design Limit is low enough for the system to maintain a margin to unacceptable fuel cladding damage during AOOs.

The P5 permissive automatically enables the Low Saturation Margin Trip when the nuclear power level is greater than or equal to 10^{-5}%. The P5 permissive also automatically disables the Low Saturation Margin Trip below this power.

6. RCS Loop Flow Rate

This function initiates a reactor trip and is inhibited below a certain level of nuclear power under which the protection is not necessary because DNB is no longer a risk in this condition. There are two trips:

a. Low-Low RCS Loop Flow Rate in One Loop, and
b. Low RCS Loop Flow Rate in Two Loops.

These trips protect against the following postulated accidents or AOOs:

- Decrease in heat removal by the secondary system, and
- Decrease in RCS flow rate.

The Low-Low RCS Loop Flow in One Loop Trip (6.a) requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 70% RTP:

- RCS Loop Flow sensors,
- APUs, and
- ALUs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Analytical Limit is high enough to provide an operating envelope that prevents unnecessary Low-Low Loop Flow Rate reactor trips. The Analytical Limit is low enough for the system to maintain a margin to ensure DNBR limits are met for AOOs and bounded for postulated accidents.

The P3 permissive automatically enables the Low-Low RCS Loop Flow Rate Trip (One Loop) when the nuclear power level is greater than or equal to 70% RTP. The P3 permissive also automatically disables the Low-Low RCS Loop Flow Rate Trip (One Loop) below this power.

The Low RCS Loop Flow in Two Loops Trip (6.b) requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- RCS Loop Flow sensors,
- APUs, and
- ALUs.

The Analytical Limit is high enough to provide an operating envelope that prevents unnecessary Low Loop Flow Rate reactor trips. The Analytical Limit is low enough for the system to maintain a margin to ensure DNBR limits are met for AOOs.

The P2 permissive automatically enables the Low RCS Loop Flow Rate Trip (Two Loops) when the nuclear power level is greater than or equal to 10% RTP. The P2 permissive also automatically disables the Low RCS Loop Flow Rate Trip (Two Loops) when the nuclear power level is below this power.

7. Low RCP Speed

Due to electrical transients that may affect the RCP's, a specific protection function is required. This function initiates a reactor trip and is inhibited below a low level of reactor power under which the protection is not necessary because DNB is no longer a risk.

This trip protects against the following postulated accidents or AOOs:

- Decrease in heat removal by the secondary system, and
- Decrease in RCS flow rate.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low RCP Speed Trip requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- RCP Speed Trip sensors,
- APUs, and
- ALUs.

The Analytical Limit is high enough to provide an operating envelope that prevents unnecessary Low RCP Speed reactor trips. The Analytical Limit is low enough for the system to maintain a margin to ensure DNBR limits are met for AOOs.

The P2 permissive automatically enables the Low RCP Speed Trip when the power level is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trip is also automatically disabled by permissive function P2.

8. High Neutron Flux (Intermediate Range)

This function limits the consequences of an excessive reactivity increase when the reactor is started up from a sub-critical or low power start-up condition. This trip protects against reactivity and power distribution anomalies.

The High Neutron Flux Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 when RTP is less than or equal to 10%, MODE 2, and in MODE 3 when RCSL is capable of withdrawing a RCCA or one or more RCCAs not fully inserted:

- Intermediate Range sensors,
- APUs, and
- ALUs.

The Analytical Limit is high enough to provide an operating envelope that prevents an unnecessary High Neutron Flux reactor trip. The Analytical Limit is low enough for the system to maintain a margin to unacceptable fuel cladding damage for AOOs that leads to an uncontrolled increase of the linear power density.

The P6 permissive automatically enables the High Neutron Flux Intermediate Range reactor trip when the power level is less than or equal to 10% RTP.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Low Doubling Time (Intermediate Range)

This function limits the consequences of an excessive reactivity increase when the reactor is started up from a sub-critical or low power start-up condition. This trip protects against reactivity and power distribution anomalies.

The Low Doubling Time Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 when RTP is less than or equal to 10%, MODE 2, and in MODE 3 when RCSL is capable of withdrawing a RCCA or one or more RCCAs not fully inserted:

- Intermediate Range sensors,
- APUs, and
- ALUs.

The Analytical Limit is high enough to provide an operating envelope that prevents an unnecessary Low Doubling Time reactor trip. The Analytical Limit is low enough for the system to maintain a margin to unacceptable fuel cladding damage for any postulated event that leads to an uncontrolled increase of the linear power density.

The P6 permissive automatically enables the Low Doubling Time reactor trip when the power level is less than or equal to 10% RTP.

10. Low Pressurizer Pressure

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. This trip protects against a decrease in reactor coolant inventory.

The Low Pressurizer Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- Pressurizer Pressure (Narrow Range) sensors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. The Analytical Limit is sufficiently below the full load operating value for RCS pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of an RCS depressurization.

The P2 permissive automatically enables the Low Pressurizer Pressure Trip when the power level is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trip is automatically disabled by permissive function P2.

11. High Pressurizer Pressure

In case of a RCS overpressure, a reactor trip is required in order to:

- Adapt the reactor power to the capacity of the safety systems;
- Ensure RCS integrity; and
- Avoid opening of the Pressurizer safety valves in certain primary side overpressure analyses.

This trip protects against a decrease in heat removal by the secondary system.

The High Pressurizer Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- Three Pressurizer Pressure (Narrow Range) sensors,
- Three divisions of APUs, and
- Three divisions of ALUs.

The Analytical Limit is below the nominal lift setting of the Pressurizer code safety valves, and its operation avoids the undesirable operation of these valves during normal plant operation. In the event of a complete loss of electrical load from 100% power, this setpoint ensures the reactor trip will take place, thereby limiting further heat input to the RCS and consequent pressure rise. The PSRVs may lift to prevent over pressurization of the RCS.

There are no permissives associated with this trip.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

12. High Pressurizer Level

In case of increasing Pressurizer level, a reactor trip is required in order to avoid Pressurizer over filling and to prevent the PSRVs from relieving. This trip protects against increases in reactor coolant inventory.

The High Pressurizer Level Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- Pressurizer Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The Analytical Limit is below the point where the associated transient would reach the nominal lift setting of the PSRVs, and its operation avoids the undesirable operation of these valves during normal plant operation. In the event of a CVCS malfunction, this Analytical Limit ensures a timely reactor trip will take place in order to avoid filing the pressurizer. The PSRVs may lift to prevent over pressurization of the RCS.

The P12 permissive automatically enables the High Pressurizer Level Trip when the pressure is greater than or equal to 2005 psia.

13. Low Hot Leg Pressure

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. This trip protects against a decrease in reactor coolant inventory.

The Low Hot Leg Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and in MODE 3 with the pressurizer pressure greater than or equal to 2005 psia, when the RCSL System is capable of withdrawing a RCCA, or one or more RCCAs are not fully inserted.

- Hot Leg Pressure (Wide Range) sensors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. The Analytical Limit is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of abnormal conditions.

The P12 permissive automatically enables the Low Hot Leg Pressure Trip when the pressure is greater than or equal to 2005 psia.

14. Steam Generator Pressure Drop

In case of steam or feedwater system piping failure, the affected Steam Generator (SG) depressurizes leading to a RCS cooldown and hence a reactivity transient. A reactor trip is required in order to ensure the fuel rod integrity and to adapt the reactor power to the capacity of the safety systems. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Decrease in heat removal by the secondary system.

The SG Pressure Drop Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- SG Pressure sensors,
- APUs, and
- ALUs.

In case of steam or feedwater system piping failure, the affected SG depressurizes leading to a RCS cooldown or heatup. A reactor trip is required in order to ensure the fuel rod integrity and to adapt the reactor power to the capacity of the safety systems. The Analytical Limit is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of a pipe break.

There are no permissives associated with this trip.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

15. Low SG Pressure

In case of steam or feedwater system piping failure, the affected SG depressurizes leading to a RCS cooldown and hence a criticality transient. For small breaks, the setpoint of the reactor trip on SG pressure drop may not be reached. Therefore, a reactor trip on low SG pressure is introduced in order to ensure fuel rod integrity and to adapt the reactor power to the capacity of safety systems. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Decrease in heat removal by the secondary system.

The Low SG Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and in MODE 3 with either the pressurizer pressure greater than or equal to 2005 psia, the RCSL System capable of withdrawing a RCCA, or one or more RCCAs not fully inserted:

- SG Pressure sensors,
- APUs, and
- ALUs.

In case of steam or feedwater system piping failure, the affected SG depressurizes leading to a RCS cooldown or heatup. For small breaks, the setpoint of the reactor trip on SG pressure drop may not be reached. Therefore, a reactor trip on low SG pressure is introduced in order to ensure fuel rod integrity and to adapt the reactor power to the capacity of safety systems. The Analytical Limit is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of a pipe break.

The P12 permissive automatically enables the Low SG Pressure Trip when the pressure is greater than or equal to 2005 psia.

16. High SG Pressure

In case of a loss of the main heat sink, the reactor has to be tripped in order to:

- Ensure fuel rods integrity at power;
- Adapt the reactor power to the capacity if safety systems; and
- Ensure SG integrity.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This trip protects against a decrease in heat removal by the secondary system.

The High SG Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1:

- SG Pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit is high enough to avoid spurious operation. In case of a loss of the main heat sink, the Analytical Limit is low enough to trip the reactor in order to:

- Ensure fuel rod integrity at power,
- Adapt the reactor power to the capacity of safety systems, and
- Ensure SG integrity.

There are no permissives associated with this trip.

17. Low SG Level

This trip protects the reactor from a loss of heat sink in case of SG steam/feedwater flow mismatch. This trip protects against a decrease in heat removal by the secondary system.

The Low SG Level Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- SG Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The purpose of this trip is to protect the reactor from a loss of heat sink in case of SG steam/feedwater flow mismatch. The Analytical Limit is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of a flow mismatch.

The P13 permissive automatically enables the Low SG Level Trip when the hot leg temperature is greater than or equal to 200ºF.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

18. High SG Level

This trip protects the turbine against an excessive humidity in case of a Main Feedwater (MFW) malfunction causing an increase in feedwater flow or in case of SG level increase. This reactor trip ensures core integrity during these transients since an increase in feedwater flow leads to a RCS overcooling event and hence a reactivity insertion. This trip protects against an increase in heat removal by the secondary system.

The High SG Level Trip requires the following sensors and processors to be OPERABLE in MODE 1 and in MODE 2:

– SG Level (Narrow Range) sensors
– APUs, and
– ALUs.

This reactor trip ensures core integrity during transients involving a MFW malfunction that results in an increase in feedwater flow or in case of a SG level increase. The Analytical Limit is sufficiently above the full load operating value so as not to interfere with normal plant operation, but still low enough to provide the required protection in the event of an abnormal condition.

The P13 permissive automatically enables the High SG Level Trip when the hot leg temperature is greater than or equal to 200ºF.

19. High Containment Pressure

In case of a postulated initiating event leading to water or steam discharge into the containment, a reactor trip is performed in order to ensure containment integrity and to adapt the reactor power to the capacity of the safety systems. This trip protects against the following postulated accidents or AOOs:

– Decrease in heat removal by the secondary system, and
– Decrease in reactor coolant inventory.

The High SG Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

– Containment Equipment Compartment and Containment Service Compartment pressure sensors,
– APUs, and
– ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In case of a postulated initiating event leading to water or steam discharge into the containment, a reactor trip is performed in order to ensure containment integrity and to adapt the reactor power to the capacity of the safety systems. The Analytical Limit is high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an abnormal condition. It is set low enough to initiate a reactor trip when an abnormal condition is indicated.

There are no permissives associated with this trip.

20. Manual Reactor Trip

A manual reactor trip signal can be generated from the SICS panel and the RSS. The manual trip signal from the RSS actuates a reactor trip through energizing the shunt coils of the main reactor trip breakers.

B. ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) FUNCTIONS

Each of the analyzed accidents or AOOs can be detected by one or more ESF Functions. One of the ESF Functions is the primary actuation signal for that accident. An ESF Function may be the primary actuation signal for more than one type of accident. An ESF Function may also be the secondary, or backup, actuation signal for one or more other accidents. The ESF protective functions are described below.

1. Turbine Trip on Reactor Trip

A turbine trip is required following any reactor trip in order to avoid a mismatch between primary and secondary power, which would result in excessive RCS cooldown with a potential return to critical conditions and power excursion.

The automatic Turbine Trip on Reactor Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1:

- RTCB Position Indication sensor,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A turbine trip is required following any reactor trip in order to avoid a mismatch between primary and secondary power. Such a mismatch could result in an RCS cooldown transient, with a potential inadvertent return to critical conditions. The one second time delay is an Analytical Limit.

There are no automatic permissives associated with this function.

2. Main Feedwater

a. MFW Full Load Closure on Reactor Trip (All SGs)

After a reactor trip check-back, a MFW full load isolation is required. This avoids a mismatch between primary and secondary power. Such a mismatch could result in an RCS cooldown transient, with a potential inadvertent return to critical conditions.

The automatic MFW Full Load Closure on Reactor Trip function requires four divisions of the following processors to be OPERABLE in MODE 1 and MODE 2 except when the MFW full load isolation valves are closed:

- RTCB Position Indication sensor,
- APUs, and
- ALUs.

There are no automatic permissives associated with this function.

b. MFW Full Load Closure on High SG Level (Affected SG)

In the case of an increasing SG level event, the MFW supply to the affected SG is isolated in order to avoid filling the SG, and subsequently introducing water into Main Steam line and MSRT.

This function mitigates an increase in heat removal from the secondary system.

The automatic MFW Full Load Closure on High SG Level function requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODES 2 and 3, except when all MFW full load and low load isolation valves are closed:

- SG Level sensors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The MFW Full Load Closure on High SG Level Analytical Limit is high enough to avoid spurious actuation but low enough in order to prevent water level in the SG from rising and entering the steam line.

The P13 permissive automatically enables the MFW Full Load Closure on High SG Level function when the hot leg temperature is greater than or equal to 200 ºF.

c. Startup and Shutdown Feedwater Isolation on SG Pressure Drop (All SGs)

The affected SG depressurizes for the listed events, a reactor trip is initiated on a SG pressure drop signal. Also, the Startup and Shutdown Feedwater (SSS) isolation and control valves close in all the SGs.

A complete Feedwater system isolation in the affected SG limits the coolant provided into the affected SG by the MFW/SSS. This action minimizes the mass and energy released into the containment and RCS cooldown.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Steam system piping failure, and
- Feedwater system piping failure.

The automatic SSS Feedwater Isolation on SG Pressure Drop function requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODES 2 and 3, except when all MFW full load and low load isolation valves are closed:

- SG pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit is high enough to preclude spurious operation but low enough to terminate feedwater flow before overcooling of the primary system or depletion of secondary inventory.

There are no automatic permissives associated with this function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. SSS Isolation on Low SG Pressure (All SGs)

The affected SG depressurizes in the event of a steam line or Feedwater pipe failure. In the event of a small secondary side break for which the SG pressure drop signal is never reached, this function also isolates the SSS supply to the affected SG. This action minimizes the mass and energy released into the containment.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Steam system piping failure, and
- Feedwater system piping failure.

The automatic SSS Feedwater Isolation on Low SG Pressure function is required to be OPERABLE in:

- MODES 1,
- MODE 2, except when all MFW low load isolation valves are closed, and
- MODE 3 when the pressurizer pressure is greater than or equal to 2005 psia, except when all MFW low load isolation valves are closed.

The automatic SSS Feedwater Isolation on Low SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE:

- SG pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit is high enough to preclude spurious operation but low enough to terminate feedwater flow before overcooling of the primary system or depletion of secondary inventory.

The P12 permissive automatically enables the SSS Isolation on Low SG Pressure function when the pressurizer pressure is greater than 2005 psia.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

e. SSS Isolation on High SG Level for Period of Time (Affected SGs)

During an increase in SG level after a reactor trip, the SSS systems are isolated in the affected SG in order to avoid the SG filling up and thus carryover of water into Main Steam line and subsequent water discharge by MSRT. This function mitigates Increase in Feedwater flow.

The automatic SSS Isolation on High SG Level for Period of Time function requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODES 2 and 3, except when all MFW low load isolation valves are closed:

- RTCB Position Indication,
- SG Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The SSS Isolation on High SG Level for Period of Time Analytical Limit is high enough to avoid spurious actuation but low enough in order to prevent water level in the SGs from rising and entering the steam lines.

The P13 permissive automatically enables the SSS Isolation on High SG Level for Period of Time function when the hot leg temperature is greater than 200 ºF.

3. Safety Injection System Actuation

a. Low Pressurizer Pressure

In the event of a decrease in RCS water inventory, the makeup is supplied by the Medium Head Safety Injection (MHSI) in the high pressure phase of the event and the Low Head Safety Injection (LHSI) in the low pressure phase. For a potential overcooling event, the reactivity insertion is limited by the boron injection via the MHSI. Even if the boron injection is not required, MHSI injection is needed to stabilize the RCS pressure.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Safety Injection System (SIS) Actuation function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- MSLB,
- Feedwater Line Break,
- Inadvertent opening of a pressurizer pilot operated safety valve,
- Small break LOCA,
- Steam system piping failure, and
- Large break LOCA.

The automatic SIS Actuation on Low Pressurizer Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3 with the pressurizer pressure greater than or equal to 2005 psia:

- Three Pressurizer Pressure (Narrow Range) sensors,
- Three divisions of APUs, and
- Three divisions of ALUs.

The Analytical Limit for this function is set below the full load operating value for RCS pressure so as not to interfere with normal plant operation. However, the Analytical Limit is high enough to provide an SIS actuation during an RCS depressurization.

The P12 permissive automatically enables the SIS Actuation on Low Pressurizer Pressure function when the pressurizer pressure is greater than or equal to 2005 psia.

The capability for manual initiation of the SIS is provided to the operator in the MCR. This manual initiation starts the four trains of SI. Four manual initiation controls are provided, any two of which will start the four SIS trains.

b. Low Delta P_{sat}

This function ensures SIS actuation in the hot and cold shutdown conditions with LHSI / Residual Heat Removal (RHR) in operation and at least one RCP operating.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This function mitigates the following postulated accidents or AOOs:

- Small break LOCA,
- Large break LOCA,
- Spurious opening of one Main Steam relief or safety valve,
- Inadvertent opening of a pressurizer pilot operated safety valve,
- Excessive increase in secondary steam flow, and
- MSLB.

The automatic SIS Actuation on Low Delta P_{sa} function requires four divisions of the following sensors and processors:

- Hot Leg Pressure (Wide Range) sensors,
- Hot Leg Temperature (Wide Range) sensors,
- APUs, and
- ALUs.

These sensors and processors are required to be OPERABLE in MODE 3 when Trip/Actuation Function B.3.a, SIS Actuation on Low Pressurizer Pressure, is disabled.

This function ensures SIS actuation in the hot and cold shutdown conditions with LHSI/RHR in operation and at least one of the RCPs are operating.

The Analytical Limit for the Low Delta P_{sat} function is high enough to avoid spurious operation but low enough to maintain core coverage in the event of an RCS pipe break.

The P12 permissive automatically enables the SIS Actuation on Low Delta P_{sat} function when the pressurizer pressure is less than or equal to 2005 psia. The P15 permissive automatically enables the SIS Actuation on Low Delta P_{sat} function when at least two RCPs are running, the hot leg pressure is greater than or equal to 464 psia, and when the hot leg temperature is greater than or equal to 356ºF.

The capability for manual initiation of the SIS is provided to the operator in the MCR. This manual initiation starts the four trains of SI. Four manual initiation controls are provided, any two of which will start the four SIS trains.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. RCP Trip on Low Delta-Pressure across the RCP with SIS Actuation

In case of LOCA in combination with a SIS actuation, the RCPs are tripped to prevent their operation in scenarios where timing of the pump trip is related to maintaining core cooling.

This function mitigates the following postulated accidents or AOOs:

- Inadvertent opening of a PSRV, and
- Small break LOCA.

The automatic RCP Trip on Low Delta-Pressure across RCP with SIS Actuation function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- RCP Delta-Pressure sensors,
- RCP Current sensors,
- APUs, and
- ALUs.

The Analytical Limit for the RCP Trip on Low Delta-Pressure across RCP with SIS Actuation function is high enough to avoid spurious operation but low enough to ensure core cooling is maintained.

There are no automatic permissives associated with this function.

5. Partial Cooldown on SIS Actuation

The partial cooldown consists of lowering the MSRT setpoint down to allow depressurization of the RCS by heat removal of the SGs. This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- MSLB,
- Inadvertent opening of a Pressurizer pilot operated safety valve, and
- Small break LOCA.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic Partial Cooldown on SIS Actuation function requires four divisions of the following processors to be OPERABLE in MODES 1, 2, and 3:

- APUs, and
- ALUs.

The LTSP for the Partial Cooldown Actuation on SIS Actuation function is set high enough to avoid spurious operation but low enough to ensure adequate flow from the MHSI pumps to maintain core cooling.

The P14 permissive automatically enables the Partial Cooldown on SIS Actuation function when the hot leg pressure is greater than or equal to 464 psia and the hot leg temperature is greater than or equal to 356 ºF.

6. <u>Emergency Feedwater System</u>

a. Actuation on Low-Low SG Level (All SGs)

In case of loss of MFW, the Emergency Feedwater System (EFWS) is actuated to remove residual heat via secondary side. With an EFWS actuation signal, SG blowdown is also isolated to conserve SG inventory. This function mitigates the following postulated accidents or AOOs:

- Loss of normal feedwater flow,
- Feedwater system piping failure, and
- LOOP.

The automatic EFWS Actuation on Low-Low SG Level function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2 and 3:

- SG level (Wide Range) sensors,
- APUs, and
- ALUs.

This function ensures heat is removed from the primary system through the SGs in the event of a loss of MFW or feedwater line break, as indicated by low SG level. The Analytical Limit is high enough to provide an operating envelope that prevents unnecessary actuations but low enough to ensure sufficient make-up is provided to the SGs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The P13 permissive automatically enables the EFWS Actuation on Low-Low SG Level function when the hot leg temperature is greater than or equal to 200ºF.

b. Actuation on LOOP and SIS Actuation (All SGs)

The LOOP results in a trip of the turbine, RCPs, and MFW pumps. The MFW and SSS supply cut off leads to a decrease in secondary side heat removal and the primary flow coast down further reduces the capacity of the primary coolant to remove heat from the core. As a result, primary and secondary pressures and temperatures increase. The heat is removed via MSRT and EFWS. With an EFWS actuation signal, SG blowdown is also isolated to conserve SG inventory.

This function mitigates the consequences of a Small Break LOCA.

The automatic EFWS Actuation on LOOP and SIS function requires four divisions of the following processors to be OPERABLE in MODES 1 and 2:

- 6.9 kV Bus Voltage sensors,
- APUs, and
- ALUs.

This function ensures heat is removed from the primary system through the SGs in the event of a LOCA concurrent with a LOOP.

There are no automatic permissives associated with this function.

c. Isolation on High SG Level (Affected SG)

In the case of an increasing SG level event, the EFWS supply to the affected SG is isolated in order to avoid filling the SG, and subsequently introducing water into Main Steam line and MSRT. This function precludes overfilling of the SG.

The automatic EFWS Isolation on High SG Level function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2 and 3:

- SG level (Wide Range) sensors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This function ensures the SGs are not overfilled, which could allow radioactive water to be discharged through the MSRTs. The Design Limit is high enough to provide an operating envelope that prevents unnecessary isolations but low enough to ensure the SGs are not over-filled.

The P13 permissive automatically enables the EFWS Isolation on High SG Level function when the hot leg temperature is greater than or equal to 200 ºF.

7. Main Steam Relief Train

a. Actuation on High SG Pressure

In the event of a loss of the secondary side heat sink, the residual heat is removed through the steam relief valves to the atmosphere. This is done by the MSRT. The MSRT also ensures SG overpressure protection, minimizes the actuation of the Main Steam Safety Valves (MSSVs), which reduces the risk of a stuck open safety relief valve.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Total loss of load and/or turbine trip
- Loss of main heat sink (condenser),
- Inadvertent closure of a Main Steam Isolation Valve (MSIV),
- MSLB,
- RCP seizure (locked rotor) or RCP shaft break., and
- Feedwater system piping failure.

The automatic MSRT Actuation on High SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- SG Pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit for the MSRT Actuation on High SG Pressure function is set high enough to avoid spurious operation and low enough to open and relieve SG pressure before over pressurization limits are reached.

There are no automatic permissives associated with this function.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Isolation on Low SG Pressure

The Main Steam Relief Isolation Valves (MSRIVs) are opened during events in order to control pressure in the SGs. In order to prevent a stuck open Main Steam Relief Control Valve from causing an RCS cooldown and a risk of return to critical conditions, the MSRT is isolated.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Loss of main heat sink (condenser),
- Inadvertent Opening of SG Safety or Relief Valve, and
- MSLB.

The automatic MSRT Isolation on Low SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3 with the pressurizer pressure is greater than or equal to 2005 psia:

- SG pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit for the MSRT Isolation on Low SG Pressure function is set low enough to avoid spurious operation and high enough to limit the rate of RCS cooldown.

The P12 permissive automatically enables the MSRT Isolation on Low SG Pressure function when the pressure is greater than or equal to 2005 psia.

8. <u>MSIV Closure</u>

a. Closure on SG Pressure Drop (All SGs)

In case of a secondary side Steam Line or Feedwater system pipe break, the affected SG depressurizes. This function isolates all four SGs in order to:

- Prevent draining of unaffected SG,
- Limit return to criticality conditions due to a overcooling transient,
- Limit the release of radioactivity, and
- Limit mass and energy releases into the containment.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Spurious opening of one SG safety or relief valve,
- Steam system piping failure, and
- Feedwater system piping failure.

The automatic MSIV Closure on SG Pressure Drop function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- SG Pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit for the MSIV Closure on SG Pressure Drop function is high enough to avoid SG pressure fluctuations during normal operation and low enough to isolate a SG and limit the blowdown to the value assumed in the safety analysis.

There are no automatic permissives associated with this function.

b. Closure on Low SG Pressure (All SGs)

For most Main Steam Line or Feedwater pipe breaks, the affected SG depressurizes. For small breaks, the setpoint for MSIV closure on SG pressure drop may not be reached. This function isolates all four SG on the main steam side in the event of a secondary side break in order to:

- Prevent draining of unaffected SGs,
- Limit the return to critical conditions due to a overcooling transient,
- Limit the release of radioactivity, and
- Limit mass and energy releases into the containment.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Spurious opening of one SG safety or relief valve,
- Steam system piping failure, and
- Feedwater system piping failure.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic MSIV Closure on Low SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3, except when all MSIVs are closed:

- SG pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit for the MSIV Closure on Low SG Pressure function is high enough to avoid SG pressure fluctuations during normal operation and low enough to isolate a SG and limit the blowdown to the value assumed in the safety analysis.

The P12 permissive automatically enables the MSIV Closure on Low SG Pressure function when the pressurizer pressure is greater than or equal to 2005 psia.

9. <u>Containment Isolation</u>

a. Isolation (Stage 1) on High Containment Pressure

In case of a LOCA, the containment has to be isolated in order to prevent release of radioactivity to the environment. Safeguards Building HVAC is also reconfigured to process air through High Efficiency Particulate Air (HEPA) filters to ensure 10 CFR 50.34 and 10 CFR 100.21 limits are not exceeded.

The automatic Stage 1 Containment Isolation on High Containment Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- Containment Service Compartment Pressure monitors,
- Containment Equipment Compartment Pressure monitors,
- APUs, and
- ALUs.

The Analytical Limit for the Stage 1 Containment Isolation on High Containment Pressure function is high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 and 10 CFR 100.21 limits.

There are no automatic permissives associated with this function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Isolation (Stage 1) on SIS Actuation

In case of the listed events, the containment has to be isolated in order to prevent release of radioactivity to the environment. Safeguards Building HVAC is also reconfigured to process air through HEPA filters to ensure 10 CFR 50.34 and 10 CFR 100.21 limits are not exceeded.

This function mitigates the following postulated accidents or AOOs:

- Inadvertent opening of a pressurizer pilot operated safety valve, and
- LOCA.

The automatic Stage 1 Containment Isolation on SIS Actuation function requires four divisions of the following processors to be OPERABLE in MODES 1, 2, 3, and 4:

- APUs, and
- ALUs.

There are no automatic permissives associated with this function.

c. Isolation (Stage 2) on High-High Containment Pressure

In case of a LOCA, the containment has to be isolated in order to prevent release of radioactivity to the environment.

This function mitigates the following postulated accidents or AOOs:

- Inadvertent opening of a pressurizer pilot operated safety valve, and
- LOCA.

The automatic Stage 2 Containment Isolation on High-High Containment Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- Containment Service Compartment Pressure monitors,
- Containment Equipment Compartment Pressure monitors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Design Limit for the Stage 2 Containment Isolation on High-High Containment Pressure function is high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 and 10 CFR 100.21 limits.

There are no automatic permissives associated with this function.

d. Isolation (Stage 1) on High Containment Radiation

In case of a significant release of radioactivity into the containment, the containment is isolated to ensure 10 CFR 50.34 and 10 CFR 100.21 limits are not exceeded.

This function mitigates the following postulated accidents or AOOs:

- Rod ejections,
- LOCA,
- MSLB inside containment, and
- Feedwater line break inside containment.

The automatic Stage 1 Containment Isolation on High Containment Radiation function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, 3, and 4:

- Containment High Range radiation monitors,
- APUs, and
- ALUs.

The Design Limit for the Stage 1 Containment Isolation on High Containment Radiation function is high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 and 10 CFR 100.21 limits.

There are no automatic permissives associated with this function.

10. Emergency Diesel Generator

a. Start on Degraded Grid Voltage

Following the detection of degraded voltage for a period of time on one 6.9 kV bus, the EDG associated with that bus is automatically started. This function mitigates a LOOP, which is assumed to occur independently or concurrently with postulated accidents and AOOs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic EDG Start on Degraded Grid Voltage requires four divisions of the following processors to be OPERABLE in MODES 1, 2, 3, and 4 or when the associated EDG is required to be OPERABLE in accordance with LCO 3.8.2, “AC Sources - Shutdown”:

- 6.9 kV voltage sensors,
- APUs, and
- ALUs.

This function ensures AC Power is available to mitigate a postulated concurrent design basis event.

The Design Limit for the EDG Start on Degraded Grid Voltage is high enough to avoid spurious operation but low enough to ensure that power is provided to ESF functions in the time-frame assumed in the accident analyses.

There are no automatic permissives associated with this function.

b. Start on LOOP

Following a LOOP on one 6.9 kV bus, the EDG associated with that bus is automatically started. This function mitigates a LOOP, which is assumed to occur independently or concurrently with postulated accidents and AOOs.

The automatic EDG Start on LOOP requires four divisions of the following processors to be OPERABLE in MODES 1, 2, 3, and 4 or when the associated EDG is required to be OPERABLE in accordance with LCO 3.8.2, “AC Sources - Shutdown”:

- 6.9 kV voltage sensors,
- APUs, and
- ALUs.

This function ensures AC Power is available to mitigate a postulated concurrent design basis event.

The Design Limit for the EDG Start on LOOP is high enough to avoid spurious operation but low enough to ensure that power is provided to ESF functions in the time-frame assumed in the accident analyses.

There are no automatic permissives associated with this function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. <u>Chemical and Volume Control System Charging Line Isolation</u>

a. Isolation on High-High Pressurizer Level

The isolation of the CVCS Charging Line on High-High Pressurizer Level is required to avoid filling of the pressurizer and subsequent water overflow through the safety valves.

This function protects against a CVCS malfunction that causes an increase in RCS water inventory.

The automatic CVCS Charging Line Isolation on High-High Pressurizer Level function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- Pressurizer Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The Analytical Limit is low enough to initiate appropriate mitigative actions in time to prevent the pressurizer from overfilling during the CVCS Malfunction event that may increases RCS inventory, but high enough to prevent spurious operations.

The P17 permissive automatically disables the CVCS Charging Line Isolation on High-High Pressurizer Level function when the cold leg temperature is less than or equal to 248 ºF.

b. Isolation on ADM - Shutdown Condition (RCP not operating)

The ADM function in the Shutdown Condition mitigates a dilution event where no RCPs are in operation. This function ensures that:

- The dilution is stopped when the protection is actuated, and
- The core remains sub-critical.

The automatic CVCS Charging Line Isolation on ADM - Shutdown Condition (RCP not operating) function is required to be OPERABLE in:

- MODES 5, with two or less RCPs in operation, and
- MODES 6.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic CVCS Charging Line Isolation on ADM - Shutdown Condition (RCP not operating) function requires the following sensors and processors:

- Boron Concentration - CVCS Charging Line sensors (4 divisions),
- Boron Temperature - CVCS Charging Line sensors (4 divisions),
- APUs (4 divisions), and
- ALUs (Divisions 1 and 4).

The Analytical Limit is high enough to provide an operating envelope that prevents unnecessary isolations but low enough to mitigate a dilution event in the shutdown condition where the RCPs are not in operation.

This function is required to be accompanied by Permissive P7, which represents a RCP speed shutdown condition, or an ATWS signal.

c. Isolation on ADM - Standard Shutdown Conditions

This function mitigates a homogeneous dilution event in the standard shutdown states where the RCPs are in operation. This function ensures that:

- The dilution is stopped when the protection is actuated, and
- The core remains sub-critical.

The automatic CVCS Charging Line Isolation on ADM - Standard Shutdown Conditions function is required to be OPERABLE in:

- MODES 3, with thee or more RCPs in operation,
- MODES 4, with thee or more RCPs in operation, and
- MODES 5, with thee or more RCPs in operation.

The automatic CVCS Charging Line Isolation on ADM - Standard Shutdown Conditions function requires the following sensors and processors:

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Boron Concentration - CVCS Charging Line sensors (4 divisions),
- Boron Temperature - CVCS Charging Line sensors (4 divisions),
- CVCS Charging Line Flow sensors (4 divisions),
- Cold Leg Temperature (Wide Range) sensors (4 divisions),
- APUs (4 divisions), and
- ALUs (Divisions 1 and 4).

The Analytical Limit is high enough to provide an operating envelope that prevents unnecessary isolations but low enough to mitigate a dilution event in the shutdown condition where the RCPs are in operation.

This function is required to be accompanied by a permissive signal, P8, which represents a reactor shutdown condition as indicated by RCCA position indication and disabled by the Permissive P7, which represents a RCP shutdown condition.

12.a and 12.b. PSRV Actuation - First and Second Valve

The integrity of the reactor pressure vessel must be ensured under all plant conditions. At low coolant temperature, the cylindrical part of the vessel could fail by brittle fracture before the design pressure of the RCS is reached. Therefore the low-temperature overpressure protection (LTOP) is ensured by opening of the PSRVs.

This function mitigates a low temperature overpressure event.

The automatic PSRVs Actuation function requires four divisions of the following processors to be OPERABLE when the PSRVs are required to be OPERABLE by LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)"

- Hot Leg Pressure (Wide Range) sensors,
- APUs, and
- ALUs.

The Analytical Limits for the PSRV Actuation function are high enough to prevent spurious operation but low enough to prevent RCS over pressurization.

The P17 permissive automatically enables the PSRV Actuation function when the cold leg temperature is less than or equal to 248º F.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

13. Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity

In case of a significant release of radioactivity, the Control Room HVAC is reconfigured to ensure 10 CFR 50.34 limits are not exceeded.

This function mitigates the following postulated accidents or AOOs:

- Rod ejections,
- LOCA, and
- Line breaks outside containment.

The automatic Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, 3, 4, 5, 6, and during the movement of irradiated fuel assemblies:

- Control Room HVAC Intake Activity radiation monitors,
- APUs, and
- ALUs.

The Design Limit for the Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity function is high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 limits.

There are no automatic permissives associated with this function.

C. PROTECTION SYSTEM PERMISSIVES

Protection System permissives are provided to ensure reactor trips and ESF are in the correct configuration for the current unit status. They back up operator actions to ensure Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the permissive Functions do not need to be OPERABLE when the associated reactor trip or ESF functions are outside the applicable MODES. The automatic permissives are:

1. P2 - Flux (Power Range) Measurement Higher than First Threshold

The P2 permissive is representative of PRD neutron flux measurements higher than a low-power setpoint value. The P2 setpoint value corresponds to the value below which transients do not lead to risk of DNB (10% RTP).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The P2 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip 1.a - Low DNBR,
- Reactor Trip 1.b - Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c - Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d - Low DNBR - High Quality,
- Reactor Trip 1.e - Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 2 - High Linear Power Density,
- Reactor Trip 6.b - Low RCS Flow Rate in Two Loops,
- Reactor Trip 7 - Low RCP Speed, and
- Reactor Trip 10 - Low Pressurizer Pressure.

To generate the permissive, neutron flux measurements from the PRDs are compared to the setpoint. When two out of four measurements are greater than the setpoint, the permissive is validated. Otherwise, it is inhibited.

The value of the permissive was selected such that AOOs do not challenge the DNBR or centerline melt limits when they occur at a core power level below the permissive value.

2. <u>P3 - Flux (Power Range) Measurement Higher than Second Threshold</u>

The P3 permissive is representative of PRD neutron flux measurements higher than an intermediate power setpoint value. The P3 setpoint value corresponds to the value below which loss of one reactor coolant pump does not lead to risk of DNB (70% Nuclear Power).

The P3 permissive is utilized in Reactor Trip 6.a - Low-Low RCS Flow Rate in One Loop.

To generate the permissive, neutron flux measurements from the PRDs are compared to the setpoint. When two out of four measurements are greater than the setpoint, the permissive is validated.

The value of the permissive was selected such that AOOs and postulated accidents that consider a loss of one RCP do not challenge the DNBR limit when they occur at a core power level below the permissive value (70% RTP).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. P5 - Flux (Intermediate Range) Measurement Higher than Threshold

The P5 permissive is representative of intermediate range detector (IRD) neutron flux measurements above a low-power setpoint value. The P5 setpoint value corresponds to the boundary between the operating ranges of the source range detectors and intermediate range detectors (greater than or equal to 10^{-5}% power as shown on the IRDs).

The P5 permissive is utilized in the following reactor trips or ESF functions:

– Reactor Trip 4 - High Core Power Level, and
– Reactor Trip 5 - Low Saturation Margin.

To generate the permissive, neutron flux measurements from the IRDs are compared to the setpoint. When two out of four of the measurements are greater than the setpoint, the permissive is validated.

The value of the permissive defines the boundary between the operating range of the source range detectors and the operating range of the intermediate range detectors.

4. P6 - Thermal Core Power Higher than Threshold

The P6 permissive is representative of core thermal power above a low-power setpoint value corresponding to the boundary between the operating ranges of the IRDs and the PRDs (10% RTP).

The P6 permissive is utilized in the following reactor trips or ESF functions:

– Reactor Trip 8 - High Neutron Flux (Intermediate Range), and
– Reactor Trip 9 - Low Doubling Time (Intermediate Range).

Hot leg pressure measurements, hot leg temperature measurements, and cold leg temperature measurements are used to calculate core thermal power. These calculated core thermal power levels are compared to the setpoint. When three out of four of the calculated core thermal power levels are greater than the setpoint, the operator is prompted to manually validate the permissive.

The value of the permissive was selected at the boundary between the operating range of the intermediate range detectors and the power range detectors.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. P7 - RCP Speed Lower than Threshold

The P7 permissive defines when reactor coolant pumps (RCPs) are no longer in operation. The P7 permissive is utilized in the following reactor trips or ESF functions:

- ESF 11.b - CVCS Charging Line Isolation on ADM at Shutdown Condition (RCP not operating), and
- ESF 11.c - CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

The RCP speed measurements (one per RCP) are compared to a setpoint (91% nominal speed). When two out of four of the measurements are less than the setpoint, the permissive is validated (i.e., indicates that two or more RCPs are turned off).

The value of the permissive was selected to establish the requirements for anti-dilution mitigation in a timely manner.

6. P8 - Shutdown RCCA Position Lower than Threshold

The P8 permissive defines the shutdown state with all rods in (ARI). The P8 permissive is utilized in ESF 11.c - CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

RCCA Bottom Position Indicator sensors are acquired in four different electrical divisions. For each division, when all rods in the shutdown banks reach the lower end position, a signal is generated. When two out of four of divisions indicate all rods in, the permissive is validated.

The P8 Permissive is characteristic of a shutdown state with ARI. With an ARI condition, this permissive enables the Anti-dilution in Standard Shutdown States function and inhibits the Anti-dilution in Power Condition" function.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. P12 - Pressurizer Pressure Lower than Threshold

The P12 permissive defines the transition from hot shutdown to cold shutdown with respect to RCS pressure. The P12 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip 12 - High Pressurizer Level,
- Reactor Trip 13 - Low Hot Leg Pressure,
- Reactor Trip 15 - Low SG Pressure Trip,
- ESF 2.d - SSS Isolation on Low SG Pressure (All SGs),
- ESF 3.a - SIS Actuation on Low Pressurizer Pressure,
- ESF 3.b - SIS Actuation on Low Delta Psat,
- ESF 7.b - MSRT Isolation on Low SG Pressure
- ESF 8.b - MSIV Closure on Low SG Pressure (All SGs), and
- ESF 9.b - Containment Isolation (Stage 1) on SIS Actuation.

Pressurizer pressure measurements are compared to the P12 setpoint (2005 psia). The low SG pressure and low hot leg pressure reactor trip functions are automatically activated when the pressurizer pressure rises above the P12 permissive value.

The Permissive P12 reflects the transition from hot shutdown to cold shutdown. P12 ensures cooling by Main Steam Bypass or MSRT down to the LHSI/RHR connection temperature and to be able to depressurize the reactor coolant system to LHSI/RHR connection pressure without actuation of SIS.

8. P13 - Hot Leg Temperature Lower than Threshold

The P13 permissive defines when steam generator draining and filling operations are allowed. The P13 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip17 - Low SG Level,
- Reactor Trip 18 - High SG Level,
- ESF 2.b - MFW Full Load Closure on High SG Level (Affected SGs)
- ESF 2.e - SSS Isolation on High SG Level for Period of Time (Affected SGs),
- ESF 6.a - EFWS Actuation on Low-Low SG Level (All SGs), and
- ESF 6.c - EFWS Isolation on High SG Level (Affected SG).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Hot leg temperature (WR) measurements are compared to the P13 setpoint (200ºF).

The value of the permissive was selected in order to permit draining and filling operations during shutdown and LHSI/RHR in operation without generating protection signals.

9. P14 - Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds

The P14 permissive defines when the residual heat removal system is allowed to be connected to the RCS. The P14 permissive is utilized in ESF 5 - Partial Cooldown Actuation on SIS Actuation.

At pressures and temperatures below the setting of the P14 permissive (464 psia and 356 ºF), operation of the LHSI/RHR system is allowed.

This permissive is manually controlled.

10. P15 - RCPs Shutdown and P14

The P15 permissive defines when SI actuation due to delta Psat is disabled and SI actuation due to low loop level is enabled.

The P15 permissive is utilized in the following reactor trips or ESF functions:

- ESF 3.b - SIS Actuation on Low Delta Psat, and
- ESF 9.b - Containment Isolation (Stage 1) on SIS Actuation.

The value for Permissive P15 (50% no load current and P14 is true) represents the threshold for switching from the SIS Actuation on Low Delta P_{sat} protection to protection via the SIS Actuation on Low RCS Loop Level.

11. P17 - Cold Leg Temperature Lower than Threshold

The P17 permissive corresponds to the temperature conditions where brittle fracture protection is required. The P17 permissive is utilized in the following reactor trips or ESF functions:

- ESF 12.a - PSRV Actuation - First Valve, and
- ESF 12.b - PSRV Actuation - Second Valve.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The value for Permissive P17 is the threshold for activation of cold overpressure mitigation systems.

D. SENSORS, MANUAL ACTUATION SWITCHES, SIGNAL PROCESSORS, AND ACTUATION DEVICES

The relationship between sensors, manual actuation switches, signal processors, and actuation devices is provided below:

SENSORS

1. <u>6.9 kV Bus Voltage</u>

Three 6.9 kV Bus Voltage sensors per EDG are required to be OPERABLE in MODES 1, 2, 3, 4, and when the associated EDG is required to be OPERABLE by LCO 3.8.2. These sensors support the following functions:

- ESF 6.b: EFWS Actuation on LOOP and SIS Actuation (All SGs),
- ESF 10.a: EDG Start on Degraded Grid Voltage, and
- ESF 10.b: EDG Start on LOOP.

2. <u>Boron Concentration - CVCS Charging Line</u>

Four Boron Concentration - CVCS Charging Line sensors are required to be OPERABLE in MODES 3 and 4 with three or more RCPs in operation and in MODES 5 and 6. These sensors support the following functions:

- ESF 11.b: CVCS Charging Line Isolation on ADM at Shutdown Condition (RCP not operating), and
- ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Boron Temperature - CVCS Charging Line

Four Boron Temperature - CVCS Charging Line sensors are required to be OPERABLE in MODES 3 and 4 with three or more RCPs in operation and in MODES 5 and 6. These sensors support the following functions:

- ESF 11.b: CVCS Charging Line Isolation on ADM at Shutdown Condition (RCP not operating), and
- ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

4. CVCS Charging Line Flow

Four CVCS Charging Line Flow sensors are required to be OPERABLE in MODES 3, 4, and 5 when three or more RCPs are in operation. These sensors support ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

5. Cold Leg Temperature (Narrow Range)

Four Cold Leg Temperature (Narrow Range) sensors are required to be OPERABLE when RTP is greater than or equal to 10%. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop, and
- Permissive P6: Thermal Core Power Higher than Threshold.

6. Cold Leg Temperature (Wide Range)

Four Cold Leg Temperature (Wide Range) sensors are required to be OPERABLE in:

- MODE 1,
- MODE 2, when power is greater than or equal to 10^{-5}% as shown on the intermediate range detectors, and in
- MODES 3, 4, 5, and 6 with three or more RCPs in operation.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

These sensors support the following functions and Permissives:

- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin,
- ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions, and
- Permissive P17: Cold Leg Temperature Lower than Threshold.

7. Containment Pressure

Four Containment Equipment Compartment Containment and Service Compartment Pressure sensors per area are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- Reactor Trip 19: High Containment Pressure,
- ESF 9.a: Containment Isolation (Stage 1) on High Containment Pressure, and
- ESF 9.c: Containment Isolation (Stage 2) on High-High Containment Pressure.

8. Hot Leg Pressure (Wide Range)

Four Hot Leg Pressure (Wide Range) sensors are required to be OPERABLE in Modes 1, 2, and 3, and when the PSRVs are required to be OPERABLE per LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)." These sensors support the following functions and Permissives:

- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin,
- Reactor Trip 13: Low Hot Leg Pressure,
- ESF 3.b: SIS Actuation on Low Delta P_{sat},
- ESF 12.a: PSRV Actuation - First Valve,
- ESF 12.b: PSRV Actuation - Second Valve,
- Permissive P6: Thermal Core Power Higher than Threshold,
- Permissive P14: Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds, and
- Permissive P15: RCPs Shutdown and P14.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Hot Leg Temperature (Narrow Range)

Four Hot Leg Temperature (Narrow Range) sensors in each of four divisions (16 total) are required to be OPERABLE in MODE 1 and MODE 2 when power is greater than or equal to 10^{-5}% as shown on the intermediate range detectors. These sensors support the following functions and Permissives:

- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin, and
- Permissive P6: Thermal Core Power Higher than Threshold.

10. Hot Leg Temperature (Wide Range)

Four Hot Leg Temperature (Wide Range) sensors are required to be OPERABLE in MODE 3 when Trip/Actuation Function B.3.a, SIS Actuation on Low Pressurizer Pressure, is disabled. These sensors support the following functions and Permissives:

- ESF 3.b: SIS Actuation on Low Delta P_{sat},
- Permissive P13: Hot Leg Temperature Lower than Threshold,
- Permissive P14: Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds, and

11. Intermediate Range

Four Intermediate Range sensors are required to be OPERABLE in:

- MODE 1, when power is less than or equal to 10% RTP,
- MODE 2, and in
- MODES 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

These sensors support the following functions and Permissives:

- Reactor Trip 8: High Neutron Flux (Intermediate Range),
- Reactor Trip 9: Low Doubling Time (Intermediate Range), and
- Permissive P5: Flux (Intermediate Range) Measurement Higher than Threshold.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

12. Power Range

Two Power Range sensors per division (8 total) are required to be OPERABLE in MODES 1 and 2, and in MODE 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These sensors support the following functions and Permissives:

- Reactor Trip 3: High Neutron Flux Rate of Change,
- Permissive P2: Flux (Power Range) Measurement Higher than First Threshold, and
- Permissive P3: Flux Measurement (Power Range) Higher than Second Threshold.

13. Pressurizer Level (Narrow Range)

Four Pressurizer Level (Narrow Range) sensors are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- Reactor Trip 12: High Pressurizer Level, and
- ESF 11.a: CVCS Charging Line Isolation on High-High Pressurizer Level.

14. Pressurizer Pressure (Narrow Range)

Four Pressurizer Pressure (Narrow Range) sensors are required to be OPERABLE in MODES 1 and 2 and MODE 3 when the pressurizer pressure is less than or equal to 2005 psia. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 10: Low Pressurizer Pressure,
- Reactor Trip 11: High Pressurizer Pressure,
- ESF 3.a: SIS Actuation on Low Pressurizer Pressure, and
- Permissive P12: Pressurizer Pressure Lower than Threshold.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

15. Radiation Monitor - Containment High Range

Four Containment High Range Radiation Monitors are required to be OPERABLE in MODES 1, 2, 3, and 4. These sensors support ESF 9.d: Containment Isolation (Stage 1) on High Containment Radiation.

16. Radiation Monitor - Control Room HVAC Intake Activity

Four Control Room HVAC Intake Activity Radiation Monitors are required to be OPERABLE in MODES 1, 2, 3, 4, 5, 6, and during the movement of irradiated fuel assemblies. The monitors are not required to be OPERABLE when the associated train is in the recirculation mode. These sensors support ESF 13: Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity.

17. RCP Current

Three RCP Current sensors per RCP (12 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions and Permissives:

- ESF 4: RCP Trip on Low Delta P across RCP with SIS Actuation, and
- Permissive P15: Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds and Reactor Coolant Pumps Shutdown.

18. RCP Delta P Sensors

Two RCP Delta-Pressure sensors per pump (8 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support ESF 4: RCP Trip on Low Delta P across RCP with SIS Actuation.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

19. RCP Speed

Four RCP Speed sensors are required to be OPERABLE when RTP is greater than or equal to 10%. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 7: Low RCP Speed, and
- Permissive P7: RCP Speed Lower than Threshold.

20. RCS Loop Flow

Four RCS Loop Flow sensors per loop (16 total) are required to be OPERABLE in MODE 1 and in MODE 2 when power is greater than or equal to 10^{-5}% as shown on the intermediate range detectors. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin,
- Reactor Trip 6a: Low-Low RCS Loop Flow Rate in One Loop,
- Reactor Trip 6b: Low RCS Loop Flow Rate in Two Loops, and
- Permissive P6: Thermal Core Power Higher than Threshold.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

21. RTCB Position Indication

Four RTCB Position Indication sensors are required to be OPERABLE in MODE 1 and in MODES 2 and 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These sensors support the following functions:

- ESF 1: Turbine Trip on Reactor Trip,
- ESF 2.a: MFW Full Load Closure on Reactor Trip (All SGs), and
- ESF 2.e: MFW and SSS Isolation on High SG Level for Period of Time (Affected SGs).

22. Self-Powered Neutron Detectors

Seventy two SPNDs are required to be OPERABLE when RTP is greater than or equal to 10%. These sensors support the following functions:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop, and
- Reactor Trip 2: High Linear Power Density.

23. SG Level (Narrow Range)

Four SG Level (Narrow Range) sensors per SG (16 total) are required to be OPERABLE in MODE 1 and in MODES 2 and 3, except when all MFW isolation valves are closed. These sensors support the following functions:

- Reactor Trip 17: Low SG Level,
- Reactor Trip 18: High SG Level,
- ESF 2.b: MFW Full Load Closure on High SG Level (Affected SGs), and
- ESF 2.e: SSS Isolation on High SG Level for Period of Time (Affected SGs).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

24. SG Level (Wide Range)

Four SG Level (Wide Range) sensors per SG (16 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- ESF 6.a: EFWS Actuation on Low-Low SG Level (All SGs), and
- ESF 6.c: EFWS Isolation on High SG Level (Affected SG).

25. SG Pressure

Four SG Pressure sensors per SG (16 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- Reactor Trip 14: SG Pressure Drop,
- Reactor Trip 15: Low SG Pressure,
- Reactor Trip 16: High SG Pressure,
- ESF 2.c: SSS Isolation on SG Pressure Drop (All SGs),
- ESF 2.d: SSS Isolation on Low SG Pressure (All SGs),
- ESF 7.a: MSRT Actuation on High SG Pressure,
- ESF 7.b: MSRT Isolation on Low SG Pressure,
- ESF 8.a: MSIV Closure on SG Pressure Drop (All SGs), and
- ESF 8.b: MSIV Closure on Low SG Pressure (All SGs).

MANUAL ACTUATION SWITCHES

1. Reactor Trip

Four manual Reactor Trip switches are required to be OPERABLE in MODES 1 and 2 and in MODES 3, 4, and 5 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These sensors support all reactor trip functions.

2. SIS Actuation

Four manual SIS Actuation switches are required to be OPERABLE in MODES 1, 2, 3, and 4. These sensors support the following functions:

- ESF 3.a: SIS Actuation on Low Pressurizer Pressure,
- ESF 3.b: SIS Actuation on Low Delta P_{sat}.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. SG Isolation

Four manual SG Isolation switches per SG (16 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- ESF 2.b: MFW Full Load Closure on High SG Level (Affected SGs);
- ESF 2.c. SSS Isolation on SG Pressure Drop (All SGs);
- ESF 5: Partial Cooldown Actuation on SIS Actuation;
- ESF 6.a: EFWS Actuation on Low-Low SG Level (All SGs);
- ESF 6.c. EEFWS Isolation on High SG Level (Affected SG); and
- ESF 8.a: MSIV Closure on SG Pressure Drop (All SGs).

SIGNAL PROCESSORS

1. Remote Acquisition Units

Two RAUs per division (8 total) are required to be OPERABLE when RTP is greater than or equal to 10%. These signal processors support the following functions:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop, and
- Reactor Trip 2: High Linear Power Density.

2. Acquisition and Processing Units

Five APUs per division (20 total) are required to be OPERABLE in accordance with the supported functions as shown in Table 3.3.1-2. These signal processors support the reactor trip, ESF functions, and Permissives.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Actuation Logic Units

Four ALUs per division (16 total) are required to be OPERABLE in MODES 1, 2, 3, 4, 5, 6, and during the movement of irradiated fuel assemblies. These signal processors support the reactor trip, ESF functions and Permissives.

ACTUATION DEVICES

1. RCP Bus and Trip Breakers

Two RCP Bus and Trip Breakers per pump (8 total) are required to be OPERABLE in MODES 1, 2, 3, and 4. These actuation devices support ESF 4: RCP Trip on Low Delta P across RCP with SIS.

2. Reactor Trip Circuit Breakers

Four RTCBs are required to be OPERABLE in MODES 1 and 2 and in MODE 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These actuation devices support the reactor trip functions.

3. Reactor Trip Contactors

Twenty three sets of four Reactor Trip Contactors (92 total) are required to be OPERABLE in MODES 1 and 2 and in MODE 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These actuation devices support the reactor trip functions.

ACTIONS

The most common causes of division inoperability are outright failure or drift of the sensor sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CALIBRATION when the sensor is set up for adjustment to bring it to within specification. The SCP ensures that the divisions are performing as expected by confirming that the drift and other related errors are consistent with the supporting setpoint calculations.

BASES

ACTIONS (continued)

When the number of inoperable sensors or signal processors in a reactor trip or ESF function exceeds that specified in any related Condition, redundancy is lost and actions must be taken to restore the required redundancy.

A Note has been added to the ACTIONS. The Note has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each PS sensor, manual actuation switch, signal processor, and actuation device. The Completion Times of each inoperable sensor, manual actuation switch, signal processor, and actuation device will be tracked separately, starting from the time the Condition was entered for that sensor, manual actuation switch, signal processor, and actuation device.

A.1 and A.2

Condition A applies to the failure of one or more sensors. Condition A.1 applies only to the RTCB Position Indication sensors. If one or more of these sensors is inoperable, the inoperable sensor(s) must be placed in the tripped condition in 1 hour. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator action. Condition A.2 applies to all other PS sensors. If one or more of these sensors is inoperable, the inoperable sensor is placed in lockout in 4 hours. The 4 hour allotted timeframe is sufficient to allow the operator to take all appropriate actions for the failed sensor and still ensures that the risk involved in operating with the failure is acceptable.

B.1

Condition B applies to the failure of one or more manual actuation switches. In this condition, the minimum functional capability for manual actuation may not maintained. Restoring the manual initiation capability to OPERABLE status within 48 hours is reasonable considering the availability of automatic actuation, the low probability of an AOO or postulated accident occurring during this time, and the time necessary for repairs.

ACTIONS (continued)

C.1 and C.2

Condition C applies to one or more APUs inoperable due to the Setpoint Control Program requirements for one or more Trip/Actuation Functions not met. In this condition, the hardware is still functional. The sensors have been calibrated and the ADOTs and SOTs have checked the function from sensor to actuation device. The manual actuation capability would be unaffected. If the associated Setpoint Control Program requirements are not met for either the EDG Start on Degraded Grid Voltage or the EDG Start on LOOP (Trip/Actuation Functions B.10.a or B.10.b), Required Action C.1 directs entry into the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown." The Completion Time of 1 hour is a reasonable time to allow the operator to diagnose and potentially correct the issue that caused the noncompliance with the associated Setpoint Control Program requirements prior to entering LCO 3.8.1 or LCO 3.8.2. Restoring compliance with the Setpoint Control Program requirements within 24 hours for all other Trip/Actuation Functions is a reasonable timeframe considering the time necessary to change the setpoint parameter, load corrected software, or replace the unit. If compliance with the Setpoint Control Program requirements cannot be restored, the associated Trip/Actuation Function must be placed in lockout in the associated APU.

D.1 and D.2

Condition D applies to one or more signal processors inoperable for reasons other than Condition C. If the inoperability affects the APU associated with the EDG Start on Degraded Grid Voltage or the EDG Start on LOOP (Trip/Actuation Functions B.10.a or B.10.b), Required Action D.1 directs entry into the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown." The Completion Time of 1 hour is a reasonable time to allow the operator to diagnose and potentially correct the issue that caused the inoperability prior to entering LCO 3.8.1 or LCO 3.8.2. Restoring the Signal processor to OPERABLE status within 4 hours for all other Trip/Actuation Functions is a reasonable timeframe considering the time necessary to restore the signal processor to OPERABLE status. If the signal processor cannot be restored to OPERABLE status, the signal processor must be placed in lockout.

BASES

ACTIONS (continued)

E.1

Condition E applies to the RCP Bus and Trip Breakers, RTCBs, and Reactor Trip Contactors. With one ore more actuation devices inoperable, the actuation device must be restored to OPERABLE status within 48 hours. The Completion Time of 48 hours is reasonable considering that there are two automatic actuation divisions and the low probability of an event occurring during this interval.

F.1

If the Required Action and associated Completion Time of Condition A, B, C, D, or E or if the minimum functional capability (the value where the supported functions would not actuate during an AOO or postulated event coupled with a single failure) of the sensors, manual actuation switches, signal processors or actuation devices specified in Table 3.3.1-1 are not met, then the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE and any other specified actions must be taken.. The applicable Condition referenced in the table is sensor, manual actuation switch, signal processor, actuation device, and MODE dependent. Condition F is entered to provide for transfer to the appropriate subsequent Condition. Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1.

G.1

If Table 3.3.1-1 directs entry into Condition G, the unit must be brought to a condition in which the Low-Low RCS Loop Flow Rate in One Loop function (Trip/Actuation Function A.6.a) is not required to be OPERABLE. The allowed Completion Time of 2 hours is reasonable, based on operating experience, to reduce THERMAL POWER from full power to less than 70% in an orderly manner and without challenging unit systems.

H.1

If Table 3.3.1-1 directs entry into Condition H, the unit must be brought to a condition in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reduce THERMAL POWER from full power to less than 10% in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

I.1

If Table 3.3.1-1 directs entry into Condition I, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging unit systems.

J.1

If Table 3.3.1-1 directs entry into Condition J, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

K.1 and K.2

If Table 3.3.1-1 directs entry into Condition K, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and open the reactor trip breakers without challenging unit systems.

L.1 and L.2

If Table 3.3.1-1 directs entry into Condition L, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and then reduce the pressurizer pressure to less than 2005 psia without challenging unit systems.

ACTIONS (continued)

M.1 and M.2

If Table 3.3.1-1 directs entry into Condition M, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours to reach MODE 3 and 12 hours to reach MODE 4 is reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

N.1 and N.2

If Table 3.3.1-1 directs entry into Condition N, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours to reach MODE 3 and 36 hours to reach MODE 5 is reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

O.1

If Table 3.3.1-1 directs entry into Condition O, the Conditions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, "AC Sources - Shutdown," for the EDG made inoperable by failure of the 6.9 kV Bus Voltage sensors are required to be entered immediately. The actions of those LCOs provide adequate compensatory actions to assure unit safety.

P.1

If Table 3.3.1-1 directs entry into Condition P, the associated CVCS isolation valves are immediately declared inoperable. The actions of LCO 3.4.9, "Pressurizer," provide adequate compensatory actions to assure unit safety.

Q.1

If Table 3.3.1-1 directs entry into Condition Q, the associated PSRVs are immediately declared inoperable. The actions of LCO 3.4.10, "Pressurizer Safety Relief Valves," provide adequate compensatory actions to assure unit safety.

BASES

ACTIONS (continued)

R.1

If Table 3.3.1-1 directs entry into Condition R, both Control Room Emergency Filtration trains are immediately declared inoperable. The actions of LCO 3.7.10, "Control Room Emergency Filtration (CREF)," provide adequate compensatory actions to assure unit safety.

S.1

If Table 3.3.1-1 directs entry into Condition S, the manual Reactor Trip switches are inoperable. If the switches cannot be returned to OPERABLE status within one hour, actions must be taken to ensure all RCCAs are inserted and the reactor must be placed in a condition where the RCCA can not be withdrawn. This is accomplished by opening the reactor trip breakers. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator action.

T.1

If Table 3.3.1-1 directs entry into Condition T, the associated ALUs must be immediately declared inoperable. If the ALUs cannot be returned to OPERABLE status within one hour, actions must be taken to ensure all RCCAs are inserted and the reactor must be placed in a condition where the RCCA can not be withdrawn. This is accomplished by opening the reactor trip breakers. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator action.

SURVEILLANCE REQUIREMENTS

The SRs for any particular PS sensor, manual actuation switch, signal processor, or actuation device are found in the SR column of Table 3.3.1-1 for that sensor, manual actuation switch, signal processor, or actuation device.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The digital PS provides continual online automatic monitoring of each of the input signal in each division, perform software limit checking (signal online validation) against required acceptance criteria, and provide hardware functional validation so that a division check is continuously being performed. If any PS input signal is identified to be in a failure status, this condition is alarmed in the Control Room. As such, a periodic "channel check" is no longer necessary.

SR 3.3.1.1

SR 3.3.1.1 compares the calorimetric heat balance calculation to the power range division output every 24 hours. If the calorimetric heat balance calculation results exceed the power range division output by more than 2% RTP, the power range division is not declared inoperable, but must be adjusted. The power range division output shall be adjusted consistent with the calorimetric heat balance calculation results if the calorimetric calculation exceed the power range division output by more than + 2% RTP. If the power range division output cannot be properly adjusted, the division I is declared inoperable.

If the calorimetric is performed at part power (< 70% RTP), adjusting the power range division indication in the increasing power direction will assure a reactor trip below the safety analysis limit (< 11% RTP). Making no adjustment to the power range division in the decreasing power direction due to a part power calorimetric assures a reactor trip consistent with the safety analyses.

This allowance does not preclude making indicated power adjustments, if desired, when the calorimetric heat balance calculation is less than the power range division output. To provide close agreement between indicated power and to preserve operating margin, the power range divisions are normally adjusted when operating at or near full power during steady-state conditions. However, discretion must be exercised if the power range division output is adjusted in the decreasing power direction due to a part power calorimetric (< 70% RTP). This action may introduce a non-conservative bias at higher power levels. The cause of

BASES

SURVEILLANCE REQUIREMENTS (continued)

the potential non-conservative bias is the decreased accuracy of the calorimetric at reduced power conditions. The primary error contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement, which is typically a delta P measurement across a feedwater venturi. While the measurement uncertainty remains constant in delta P as power decreases, when translated into flow, the uncertainty increases as a square term. Thus a 1% flow error at 100% power can approach a 10% flow error at 30% RTP even though the delta P error has not changed. An evaluation of extended operation at part power conditions would conclude that it is prudent to administratively adjust the setpoint of the High Neutron Flux Rate of Change when: 1) the power range division output is adjusted in the decreasing power direction due to a part power calorimetric below 70% RTP; or 2) for a post refueling startup. The evaluation of extended operation at part power conditions would also conclude that the potential need to adjust the indication of the High Neutron Flux Rate of Change in the decreasing power direction is quite small, primarily to address operation in the intermediate range about 10% RTP to allow enabling of the High Neutron Flux Rate of Change reactor trips. Before the High Neutron Flux Rate of Change setpoint is reset, the power range division adjustment must be confirmed based on a calorimetric performed at ≥ 70% RTP.

The Note clarifies that 12 hours are allowed for performing the first Surveillance after reaching 20% RTP. A power level of 20% RTP is chosen based on plant stability, (i.e., automatic rod control capability and turbine generator synchronized to the grid). The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range division output of more than +2% RTP is not expected in any 24 hour period.

SR 3.3.1.2

Space- and time- dependent power density distribution of the U.S. EPR is accurately assessed using the SPNDs inside the core. For neutron flux measurement, incore neutron detectors are more accurate than excore neutron detectors. CALIBRATION of SPND instrumentation is performed to compensate for a decrease in SPND sensitivity during the fuel cycle and to account for peak power density factor change over the fuel cycle. The Aeroball Measurement System (AMS) assists in generating the measured relative neutron flux density in the core, which is used in

BASES

SURVEILLANCE REQUIREMENTS (continued)

conjunction with the predicted power distribution based on actual core operation to calibrate the incore SPND instrumentation. Because both the power-to-signal ratio of an SPND and the reference power distribution change with core burnup, SPND signals are matched to reference signals provided by the AMS every 15 EFPD (Ref. 7).

The Note clarifies that 12 hours are allowed for performing the first Surveillance after reaching 20% RTP. A power level of 20% RTP is chosen based on plant stability, (i.e., automatic rod control capability and turbine generator synchronized to the grid).

SR 3.3.1.3

SR 3.3.1.3 is the performance of a ADOT every 31 days. This test shall verify OPERABILITY by actuation of the Reactor Trip Circuit Breakers and Reactor Trip Contactors. The ADOT may be performed by means of any series of sequential, overlapping, or total steps.

SR 3.3.1.4

The online boron meters are a half shell design and are not in contact with the reactor coolant. The concentration of boron is measured by using the neutron absorption effect of B^{10}. The boron concentration is calculated using the measured count rate. To improve the accuracy of the measurement, the temperature of the reactor coolant at the measuring point is used to adjust the boron concentration. The temperature instruments are not included as part of this Surveillance. The frequency of the boron meter CALIBRATION is conservative considering instrument reliability.

Specification 5.5.18.a requires that the Limiting Trip Setpoint (LTSP), Allowable Value (AV), as-found tolerance (AFT), and the as-left tolerance (ALT), as well as the methodology for calculating these be in the Setpoint Control Program (SCP).

The SCP provides requirements for the calibration reset and evaluation of the performance of required divisions. As indicated in Specification 5.5.18.c.1 evaluation of division performance is required for the condition where the "as-found" setting for the division is outside its AFT, but conservative with respect to the AV. Evaluation of division performance will verify that the instrument will continue to behave in accordance with design-basis assumptions. The purpose of the assessment is to ensure confidence in the instrument performance prior to returning the instrument to service. These divisions will also be identified in the Corrective Action

BASES

SURVEILLANCE REQUIREMENTS (continued)

Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for OPERABILITY. For digital division components and Functions whose instruments are mechanical devices (e.g., devices which have an "on" or "off" output or an open/close position such as limit switches, float switches, and proximity detectors), the AFT may be identical to the ALT because drift may not be an expected error.

As indicated in Specification 5.5.18.c.2, the as-left setting for the instrument is required to be returned to within the ALT around the LTSP. Where a setpoint more conservative than the LTSP is used in plant surveillance procedures, the ALT and AFT, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the AV is maintained. If the as-left instrument setting cannot be returned to a setting within the ALT, then the instrument division shall be declared inoperable.

SR 3.3.1.5

A SOT on each required reactor trip actuation device is performed every 24 months to ensure the devices will perform their intended function when needed. A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for division OPERABILITY. The SOT shall include the verification of the accuracy and time constants of the analog input modules. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint methodology as required by the SCP.

The maximum permissible response time for analog input modules is prescribed by the process engineering of the specific application. Thus for each applicable PS function, the limiting response times will be shown to be consistent with the safety requirements.

The response time testing is performed in overlapping steps:

- Verification of time constants of the input divisions during input module tests, and

- Verification of the signal propagation time within the digital system.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The response time of the analog input divisions are tested periodically by injection of test signals in the input circuits. For this purpose, an external test computer is temporarily connected to the I&C system via permanently installed test plugs. While the input from the process is deactivated (by switching off the associated division(s) power supply), a binary input is provided to the data acquisition computers. The signal distribution to other computers is designed in the application software in the same way as for the normal measuring signals. Separate outputs are provided in the voting computers for each path. During the response time tests, the test machine connected to the I&C system generates a start signal and measures the reaction time of each signal path separately to verify that it does not exceed the worst case conditions specified for the specific system configuration. The measurements are performed a number of times to determine the statistical characteristics of each signal path.

The SOT may be performed by means of any series of sequential, overlapping, or total steps.

SR 3.3.1.6

A CALIBRATION of each PS sensor (except neutron detectors) every 24 months ensures that each instrument division is reading accurately and within tolerance. A CALIBRATION shall be the adjustment, as necessary, of the sensor output such that it responds within the necessary range and accuracy to known values of the parameter that the sensor monitors. The CALIBRATION shall encompass all devices in the division required for sensor OPERABILITY. CALIBRATION of instrument divisions with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal CALIBRATION of the remaining adjustable devices in the division. The CALIBRATION may be performed by means of any series of sequential, overlapping, or total steps.

Specification 5.5.18.a requires that the Limiting Trip Setpoint (LTSP), Allowable Value (AV), as-found tolerance (AFT), and the as-left tolerance (ALT), as well as the methodology for calculating these be in the Setpoint Control Program (SCP).

The SCP provides requirements for the calibration reset and evaluation of the performance of required divisions. As indicated in Specification 5.5.18.c.1 evaluation of division performance is required for the condition where the "as-found" setting for the division is outside its AFT, but conservative with respect to the AV. Evaluation of division performance will verify that the instrument will continue to behave in accordance with design-basis assumptions. The purpose of the assessment is to ensure

BASES

SURVEILLANCE REQUIREMENTS (continued)

confidence in the instrument performance prior to returning the instrument to service. These divisions will also be identified in the Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for OPERABILITY. For digital division components and Functions whose instruments are mechanical devices (e.g., devices which have an "on" or "off" output or an open/close position such as limit switches, float switches, and proximity detectors), the AFT may be identical to the ALT because drift may not be an expected error.

As indicated in Specification 5.5.18.c.2, the as-left setting for the instrument is required to be returned to within the ALT around the LTSP. Where a setpoint more conservative than the LTSP is used in plant surveillance procedures, the ALT and AFT, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the AV is maintained. If the as-left instrument setting cannot be returned to a setting within the ALT, then the instrument division shall be declared inoperable.

SR 3.3.1.7

The features of continuous self-monitoring of the PS system are described in Reference 8. Additional tests, which require the processor to be inoperable are not normally performed during operation. These EXTENDED SELF TESTS are performed at start-up of a computer each cycle. The startup sequence is as follows:

- Hardware basic test using the internal diagnosis monitor,
- Start-up self test of the operating system, and
- Switch over to normal operation after approximately two minutes.

Additional information is provided in Section 3 of Reference 8.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a ADOT every 31 days. This test shall verify OPERABILITY by actuation of the RCP Bus and Trip Breakers. The ADOT may be performed by means of any series of sequential, overlapping, or total steps.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.9

SR 3.3.1.9 verifies that the setpoint and permissive values have been properly loaded into the applicable APU.

REFERENCES

1. ANP-10275P, Revision 0, U.S. EPR Instrument Setpoint Methodology Topical Report, March 2007.
2. 10 CFR 100.
3. 10 CFR 50, Appendix A, GDC 21.
4. ANP-10287, Incore Trip Setpoint and Transient Methodology for U.S. EPR, November 2007.
5. FSAR Chapter 15.
6. 10 CFR 50.49.
7. ANP-10271P, Revision 0, US EPR Nuclear Incore Instrumentation Systems Report, December 2006.
8. EMF-2341(P), Revision 1, Generic Strategy for Periodic Surveillance Testing of TELEPERM XS System in U.S. Nuclear Generating Stations, March 2000.

Table B 3.3.1-1 (page 1 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
A. REACTOR TRIPS						
1. Low Departure from Nucleate Boiling Ratio (DNBR) a. Low DNBR d. High Quality	≥ 10% RTP	3	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit

Table B 3.3.1-1 (page 2 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
1. Low Departure from Nucleate Boiling Ratio (DNBR) b. Low DNBR and (Imbalance or Rod Drop) c. Variable Low DNBR and Rod Drop e. High Quality and Imbalance or Rod Drop	≥ 10% RTP	3	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit

Table B 3.3.1-1 (page 3 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
2. High Linear Power Density	≥ 10% RTP	3	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit
4. High Core Power Level	1,2[(a)]	3	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit

(a) ≥ 10-5% power on the intermediate range detectors.

Table B 3.3.1-1 (page 4 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
5. Low Saturation Margin	1,2[(a)]	3	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit

(a) ≥ 10-5% power on the intermediate range detectors.

Table B 3.3.1-1 (page 5 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
B. ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) SIGNALS						
2.e. Main Feedwater / Startup and Shutdown Feedwater Isolation on Steam Generator Level High for Period of Time (Affected Steam Generators)	1,2[(b)],3[(b)]	3	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit
3.b. ESF - Safety Injection System (SIS) Actuation on Low Delta P_{sat}	3[(c)]	3	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit

(b) Except when all MFW low load isolation valves are closed.

(c) When Trip/Actuation Function B.3.a, SIS Actuation on Low Pressurizer Pressure, is disabled.

Table B 3.3.1-1 (page 6 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
4. ESF - Reactor Coolant Pump (RCP) Trip on Low Delta P across RCP with Safety Injection System Actuation	1,2,3	3	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit
11a. ESF - Chemical and Volume Control System (CVCS) Charging Line Isolation on High-High Pressurizer Level	1,2,3	3	Pressurizer Level / Acquisition and Processing Unit	Pressurizer Level / Acquisition and Processing Unit	Pressurizer Level / Acquisition and Processing Unit	Pressurizer Level / Acquisition and Processing Unit
	1,2,3	2	Actuation Logic Unit (1 of 2)			Actuation Logic Unit (1 of 2)

Table B 3.3.1-1 (page 7 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
11b. ESF - Chemical and Volume Control System (CVCS) Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating)	5$^{(d)}$,6	3	Boron Concentration Boron Temperature / Acquisition and Processing Unit	Boron Concentration Boron Temperature / Acquisition and Processing Unit	Boron Concentration Boron Temperature / Acquisition and Processing Unit	Boron Concentration Boron Temperature / Acquisition and Processing Unit
	5$^{(d)}$,6	2	Actuation Logic Unit (1 of 2)			Actuation Logic Unit (1 of 2)
11c. ESF - Chemical and Volume Control System (CVCS) Charging Line Isolation on ADM at Standard Shutdown Conditions	3,4$^{(e)}$,5$^{(e)}$	3	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit
	3,4$^{(e)}$,5$^{(e)}$	2	Actuation Logic Unit (1 of 2)			Actuation Logic Unit (1 of 2)

(d) With two or less RCPs in operation.

(e) With three or more RCPs in operation.

B 3.3 INSTRUMENTATION

B 3.3.2 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to provide operators with information that is needed during accidents.

The OPERABILITY of PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following accidents and transients when the use of the Emergency Operating Procedures (EOPs) is required.

The PAM instruments included in Table 3.3.2-1, Postaccident Monitoring Instrumentation, are required for the following reasons:

1. Perform the diagnosis specified in the emergency operating procedures (these variables are restricted to preplanned actions for the primary success path of DBAs), e.g., loss of coolant accident (LOCA);

2. Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;

3. Provide information to indicate whether plant safety functions are being accomplished for reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity (including radioactive effluent control);

4. Provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases (i.e., fuel cladding, primary coolant pressure boundary, and containment); and

5. Enable the operator to recognize which heat transfer symptom is occurring: 1) loss of subcooling margin, 2) lack of heat transfer, 3) excessive heat transfer, and 4) Steam Generator Tube Rupture.

The PAM instrumentation is displayed through the Safety Information and Control Systems (SICS), which includes the Qualified Display System (QDS). The Safety Automation System (SAS) communication with the QDS (as part of SICS) is realized through the Monitoring and Service Interfaces (MSI), and the Panel Interfaces (PI). The PI's are part of the SICS, the MSI's are part of the SAS. The SAS also provides outputs for analog meters, illuminated buttons etc., and receives inputs from Conventional Instrumentation and Controls which is included in the SICS.

BASES

BACKGROUND (continued)

The SICS calculates a margin to saturation by using the safety grade inputs of RCS Hot Leg Pressure, RCS Hot Leg Temperature, and Incore Thermocouples. The margin, both positive and negative, is available for diagnosis of plant transients. As long as adequate subcooling margin exists, core cooling is ensured. If subcooling margin is lost, actions are required to ensure core cooling and restore adequate subcooling margin. Superheat is used for Inadequate Core Cooling (ICC) determination and initiation of more severe mitigation guidance to restore saturated and ultimately subcooled coolant conditions.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the OPERABILITY of certain Regulatory Guide 1.97 variables, so that the control room operating staff can:

- Recognize when a heat transfer symptom is occurring that would require performance of the appropriate section in the emergency operating procedures.
- Perform the diagnosis specified in the emergency operating procedures (these variables are restricted to preplanned actions for the primary success path of postulated accidents), e.g., loss of coolant accident (LOCA);
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;
- Determine whether systems important to safety are performing their intended functions for reactivity control, core cooling, maintaining reactor coolant system integrity and maintaining containment integrity,
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

BASES

APPLICABLE SAFETY ANALYSES (continued)

PAM instrumentation used to support pre-planned, manually controlled actions satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii)(C). The other PAM instrumentation that perform certain functions related to verification of key safety functions and monitoring key barriers for potential breach must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, these variables are important for reducing public risk.

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 monitors that provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that perform certain functions related to verification of key safety functions and monitoring key barriers for potential breach.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident.

LCO 3.3.2 requires two OPERABLE divisions for most Functions. Two OPERABLE divisions ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident.

Furthermore, OPERABILITY of two divisions allows for a comparison during the post accident phase to confirm the validity of displayed information.

The exception to the two division requirement is Penetration Flow Path Containment Isolation Valve (CIV) Position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

BASES

LCO (continued)

PAM variables are required to meet design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.2-1.

1. Cold Leg Temperature (Wide Range)

The key variables for monitoring core cooling are Hot Leg Temperature, Core Exit Temperature, and Steam Generator Pressure. Cold Leg Temperature provides backup temperature monitoring to Hot Leg Temperature and Core Exit Temperature when forced or verified natural circulation exists. Cold Leg Temperature is used with Hot Leg Temperature and Core Exit Temperature to verify natural circulation. Cold Leg Temperature is compared to the saturation temperature for steam generator pressure (Tsat) to determine primary to secondary loop coupling.

2. Containment Isolation Valve Position Indication

Containment isolation valve position verifies Containment isolation and is required to ensure Containment integrity in event of a LOCA.

3. Containment Pressure

Containment pressure is a key measurement used for detection of a LOCA, verification of Engineered Safety Features mitigation, and detection of a potential breach of Containment.

4. Emergency Feedwater Storage Pool Level

Emergency feedwater pool level is a key variable to ensure adequate EFW pump net positive suction head (NPSH) is satisfied.

BASES

LCO (continued)

5. Emergency Feedwater System Flow

Emergency Feedwater flow indication is required when throttling feedwater flow to the steam generators. Control of flow is required to control the rate of steam generator heat removal to maintain Reactor Coolant temperature profiles within limits for cooldown.

6. Extra Boration System Flow

The Extra Boration System flow provides verification that the appropriate system alignment has been completed. The negative reactivity additions performed by this system require verfication of correct system operation.

7. Hot Leg Injection Flow

Hot Leg Injection flow provides verification that the appropriate system alignment has been completed. Hot leg injection is required to prevent the buildup of sufficient boron concentration in the core coolant channels to impede long term core cooling.

8. Hot Leg Pressure (Wide Range)

RCS pressure is required to monitor reactor coolant integrity and assess core cooling. RCS pressure and either RCS hot leg or incore temperature is used to determine subcooling margin if the calcuation is not available.

9. Hot Leg Temperature (Wide Range)

Hot Leg Temperature is required to monitor core cooling, to verify natural circulation, and to verify primary to secondary loop coupling along with steam generator pressure. Hot Leg temperature and RCS pressure are used to determine loop subcooling margin if the calculation is not available.

BASES

LCO (continued)

10. <u>In-containment Refueling Water Storage Tank Level</u>

In-Containment storage tank level is monitored during operation to ensure that adequate pump NPSH is maintained during the recirculation phase of LOCA mitigation for long term core cooling requirements. In addition, level instrumentation is used to assess level loss due to leakage on Safety Injection piping located outside of Containment and interfacing systems (Inter-system LOCA) as well as level rise due to dilution mechanisms.

11. <u>Incore Temperature</u>

Core cooling is monitored by RCS and incore thermocouple temperatures. Loss of subcooling margin (SCM) is identified by a combination of RCS pressure and either hot leg temperature or incore thermocouple temperature depending on plant conditions, (e.g., RCPs on/off). Incore Temperature is monitored for verification and surveillance of long term core cooling and to detect potential breach of fuel cladding.

12. <u>Power Range Monitors</u>

Power Range Neutron Flux is used to verify that reactor trip has resulted in "Reactor Shutdown". Once "Reactor Shutdown" is verified following reactor trip, all subsequent EOP action is based on a shutdown reactor. Power Range indication is used during a steam generator tube rupture to determine when core power is within the Main Steam bypass capability, at which point a reactor trip can be performed without challenge to the Main Steam relief capabilities.

13. <u>Pressurizer Level</u>

Pressurizer level provides information for the operator to maintain RCS pressure and inventory control, with the exception of a few accident situations, such as large break LOCA. Pressurizer level is a key variable required to ensure proper operation of the pressurizer heaters to maintain the pressurizer in a saturated state.

BASES

LCO (continued)

14. Radiation Monitor - Containment High Range Activity

Containment high range radiation instrumentation is used to assess the potential for significant radiation releases and to provide release assessment for determining the need to invoke site emergency plans.

15. Radiation Monitor - Main Steam Line Activity

Main Steam Line radiation levels are a key variable for detection of a breach between the primary to secondary loop boundary.

16. Source Range Monitors

Source Range instrumentation is used to ensure that the reactor remains subcritical. Once "Reactor Shutdown" is verified following reactor trip, all subsequent EOP action is based on a shutdown reactor. Source range can be used to assess if a return to critical condition is approached during plant cooldown and whether mitigation efforts are required to maintain the reactor in a shutdown condition.

17. Steam Generator Level (Wide Range)

Both steam generator level and pressure are monitored to assess primary to secondary heat transfer. An upper level limit is specified to prevent moisture carryover into the steam lines which could damage control components used for controlling RCS cooldown.

18. Steam Generator Pressure

Steam Generator pressure is a key parameter used to identify upsets in heat transfer and evaluate primary-to-secondary heat transfer.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate postulated accidents. The applicable postulated accidents are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, plant conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

BASES

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.2-1. The Completion Time(s) of the inoperable division(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

When one or more Functions have one required division that is inoperable, the required inoperable division must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE division (or in the case of a Function that has only one required division, other non-Regulatory Guide 1.97 instrument divisions to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

This Required Action specifies initiation of actions in accordance with Specification 5.6.5, which requires a written report to be submitted to the Nuclear Regulatory Commission. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative Required Actions. This Required Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Required Actions are identified before a loss of functional capability condition occurs.

C.1

When one or more Functions have two required divisions inoperable (i.e., two divisions inoperable in the same Function), one division in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of alternate means to obtain the required information. Continuous operation with two required divisions inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable division of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

BASES

ACTIONS (continued)

D.1 and D.2

If the Required Action and associated Completion Time of Condition C are not met and Table 3.3.2-1 directs entry into Condition E, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

A Note at the beginning of the SR Table specifies that the following SR applies to each PAM instrumentation Function found in Table 3.3.2-1.

SR 3.3.2.1

A CALIBRATION is performed every 24 months or approximately every refueling. CALIBRATION is a complete check of the instrument division including the sensor. The Surveillance verifies the function responds to the measured parameter within the necessary range and accuracy. A Note allows exclusion of the neutron detectors from the CALIBRATION. The requirements for CALIBRATION of neutron detectors is Specified in Specification 3.3.1," Protection System and Safety Automation System".

The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of a 24 month CALIBRATION interval for the determination of the magnitude of equipment drift.

SR 3.3.2.2

A SOT on each Safety Information and Control System performing the PAM functions listed in Table 3.3.2-1 is performed every 24 months to ensure the entire division will perform its intended function when needed. A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for division OPERABILITY. The SOT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for division OPERABILITY such that the setpoints are within the necessary range and accuracy. The SOT may be performed by means of any series of sequential, overlapping, or total steps.

BASES

REFERENCES

1. NUREG 0737, Supplement 1.

B 3.3 INSTRUMENTATION

B 3.3.3 Remote Shutdown System (RSS)

BASES

BACKGROUND

The RSS provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as Hot Standby (MODE 3). With the unit in MODE 3, the Emergency Feedwater (EFW) System and Main Steam Relief Train (MSRT) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the EFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allow extended operation in MODE 3.

The RSS contains Human Machine Interface (HMI) workstations necessary to bring the plant to and maintain it in a safe shutdown state. The HMI (control) functions of the RSS are isolated as long as the Main Control Room (MCR) is available. The HMI workstations will continue to display all parameters available on each workstation while the control functions are isolated. These workstations contain Process Information and Control System (PICS) equipment, Safety Information and Control System (SICS) equipment, and select communication equipment.

In the event that the control room becomes inaccessible, the operators can establish control at the remote shutdown panel and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are required to be located at the remote shutdown panel. Some controls and transfer switches may be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the RSS control and instrumentation Functions ensures that there is sufficient information available on selected plant parameters to bring the plant to, and maintain it in, MODE 3 should the control room become inaccessible.

BASES

APPLICABLE SAFETY ANALYSES

The RSS is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down the plant and maintain it in a safe condition in MODE 3.

The criteria governing the design and the specific system requirements of the RSS are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

The RSS satisfies Criterion 4 of 10 CFR 50.36(d)(2)(ii).

LCO

The RSS LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the plant in MODE 3 from a location other than the control room. The instrumentation and controls required are listed in Table B 3.3.3-1.

The controls, instrumentation, and transfer switches necessary to reach MODE 3 are those required for:

- Reactivity Control (initial and long term),
- Reactor Coolant Make-up
- RCS Pressure Control,
- Decay Heat Removal, and
- Safety support systems for the above Functions, as well as service water, component cooling water, and onsite power including the Emergency Diesel Generators.

The systems are controlled by the Safety Automation System (LCO 3.3.1, “Protection System and Safety Automation System”).

A Function of a RSS is OPERABLE if all instruments and controls needed to support the Remote Shutdown System Function are OPERABLE.

The remote shutdown instrument and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments and control circuits will be OPERABLE if unit conditions require that the RSS be placed in operation.

BASES

APPLICABILITY

The RSS LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the unit is already subcritical and in the condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument control Functions if control room instruments or control become unavailable.

ACTIONS

A RSS division is inoperable when each Function is not accomplished by at least one designated RSS division that satisfies the OPERABILITY criteria for the division's Function. These criteria are outlined in the LCO section of the Bases.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Time(s) of the inoperable division(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A addresses the situation where one or more functions of the RSS are inoperable. This includes the control and transfer switches for any required Function.

The Required Action is to restore the divisions to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

SR 3.3.3.1 verifies that each required RSS transfer switch and control circuit performs its intended function. This verification is performed from the reactor shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. Displays in the MCR and RSS contain real-time plant data prior to, during, and after control transfer from one station to the other. The RSS data is populated from the same information busses that supply data to the MCR. During the time control is transferred from the MCR to the RSS or vice versa, the operator will have seamless transfer of control and data will not be interrupted. The operators will have an indication via the control system that RSS control has been established. This will ensure that if the control room becomes inaccessible, the plant can be brought to and maintained in MODE 3 from the reactor shutdown panel and the local control stations. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience demonstrates that RSS control usually pass the Surveillance when performed at a Frequency of once every 24 months.

SR 3.3.3.2

A CALIBRATION of each instrument display function on the RSS every 24 months ensures that each instrument division is reading accurately and within tolerance. A CALIBRATION is a complete check of the instrument division, including the sensor. The test verifies that the division responds to the measured parameter within the necessary range and accuracy. CALIBRATION leaves the division adjusted to account for instrument drift to ensure that the division remains operational between successive tests.

SR 3.3.3.3

A SOT on each division performing the RSS functions is performed every 24 months to ensure the entire division will perform its intended function when needed. A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for division OPERABILITY. The SOT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for division OPERABILITY such that the setpoints are within the necessary range and accuracy. The SOT may be performed by means of any series of sequential, overlapping, or total steps.

BASES

REFERENCES	1.	10 CFR 50, Appendix A, GDC 19.

Table B 3.3.3-1 (page 1 of 2)
Remote Shutdown System Instrumentation and Controls

FUNCTION / INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF FUNCTIONS
Source Range Neutron Flux	1
Control Rod Drive Mechanism (CRDM) Bottom Position Indications	1 per CRDM
Reactor Trip Breakers	1 per trip breaker
Reactor Coolant Pump Trip	1 per pump
RCS Hot Leg Pressure Wide Range	1 per loop
RCS Hot Leg Temperature (Wide Range)	1 per loop
RCS Cold Leg Temperature (Wide Range)	1 per loop
Pressurizer Pressure	1
Pressurizer Pressure Setpoint Reset	1
Low Temperature Overpressure Alarm	1
Pressurizer Level	1
Pressurizer Level Variable Setpoints	1
Pressurizer Safety Relief Valves (incl. Actuators and Position Sensors	1 per valve
Steam Generator Pressure	1 per loop
Steam Generator Pressure Variable Setpoints	1 per loop
Steam Generator Pressure Setpoint Reset	1 per loop
Steam Generator (Wide Range) Levels	1 per loop
Main Steam Isolation Valves	1 per valve
Main Steam Relief Isolation Valves	1 per valve
Main Steam Relief Control Valves	1 per valve
In Containment Refueling Water Storage Tank (IRWST) Level	1
Low Head Safety Injection (LHSI) Pumps	1 per pump
Residual Heat Removal (RHR) Heat Exchanger Main Control Valves	1 per loop
RHR Heat Exchanger Bypass Control Valves	1 per loop
RHR Heat Exchanger Inlet Temperatures	1 per loop
RHR Heat Exchanger Outlet Temperatures	1 per loop
RHR Suction Line Isolation Valves	1 per loop
RHR Suction Line Isolation Valve Interlock Status	1 per loop
RHR Warm-Up / Conditioning Valves	1 per loop
Essential Service Water Pumps	1 per loop
Component Cooling Water (CCW) Pumps	1 per pump
CCW Surge Tank Level	1 per tank
Emergency Diesel Generator (EDG)	1 per EDG
CVCS Letdown Isolation Valves	1 per valve
EBS Boric Acid Storage Tank Levels	1
EBS Injection Line Isolation Valves	1 per valve
EBS Pumps	1 per pump

Table B 3.3.3-1 (page 2 of 2)
Remote Shutdown System Instrumentation and Controls

FUNCTION / INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF FUNCTIONS
EBS Containment Isolation Valves	1 per valve
Emergency Feedwater Pumps	1 per pump
Emergency Feedwater Pool Levels (WR)	1 per pool
P12 Permissive	1
P14 Permissive	1
P15 Permissive	1
P17 Permissive	1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and total flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and total flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS average coolant temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average coolant temperature will cause the core to approach DNB limits.

The RCS total flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS total flow rate limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS total flow rate will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.4, "AXIAL OFFSET (AO)," and LCO 3.2.5, "AZIMUTHAL POWER IMBALANCE (API)."

BASES

APPLICABLE SAFETY ANALYSES (continued)

The pressurizer pressure limit and RCS average coolant temperature limit specified in the COLR correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty.

The RCS Pressure, Temperature, and Flow DNB limits satisfy Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables – RCS pressurizer pressure, RCS average coolant temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators.

The numerical values for RCS pressurizer pressure, RCS average coolant temperature, and RCS total flow rate specified in the COLR are given for the measurement location and have been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on RCS pressurizer pressure, RCS average coolant temperature, and RCS totals flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on RCS pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

BASES

ACTIONS

A.1

With one or more of these parameters not within limits, action must be taken to restore parameter(s) in order to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Actions of A.1 are not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply.

To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Verification that RCS pressurizer pressure is within the limit specified in the COLR ensures that the initial conditions of the safety analysis are met. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Verification that RCS average coolant temperature is within the limit specified in the COLR ensures that the initial conditions of the safety analysis are met. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 24 months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Frequency of 24 months reflects the importance of verifying flow and has been shown by operating experience to be acceptable.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ≥ 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP.

REFERENCES

1. Chapter 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

APPLICABLE SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

BASES

APPLICABLE SAFETY ANALYSES (continued)

All low power safety analyses assume initial RCS loop temperatures greater than or equal to the HZP temperature of 578°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 10°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than the HZP temperature assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with $k_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with $k_{eff} < 1.0$ in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.2.1 RCS loop average temperature is required to be verified at or above 568°F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.
REFERENCES	1. Chapter 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BAKCGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation.

The Pressure Temperature Limit Report (PTLR) contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

BASES

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the Pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T Limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the Pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.2, "RCS Minimum Temperature for Criticality," and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and

BASES

APPLICABILITY (continued)

maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed within 72 hours. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

BASES

ACTIONS (continued)

B.1 and B.2

If any Required Action and associated Completion Time of Condition A is not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of time or a sufficiently severe event resulted in a determination that the RCS is or may be unacceptable for continued operation. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 370 psig within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2, and C3

Action must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis. Action must also be initiated to reduce RCS pressure to less than 370 psig to reduce the stress.

The immediate Completion Times reflect the urgency of initiating action to restore the parameters to within the analyzed range and reducing RCS pressure. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

BASES

ACTIONS (continued)

Besides restoring operation within limits and reducing RCS pressure, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.3 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration and reduction of RCS pressure alone per Required Actions C.1 and C.2 are insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. ANP-10283, Rev. 0, "U.S. EPR Pressure - Temperature Limits Methodology for RCS Heatup and Cooldown"

2. 10 CFR 50, Appendix G.

3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

BASES

REFERENCES (continued)

4. ASTM E 185-82, July 1982.

5. 10 CFR 50, Appendix H.

6. Regulatory Guide 1.99, Revision 2, May 1988.

7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The reactor coolant is circulated through four loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming four RCS loops are in operation. The majority of the plant

BASES

APPLICABLE SAFETY ANALYSES (continued)

safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the four pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the four RCS loop operation. For four RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 100.5% RTP. This is the design overpower condition for four RCS loop operation. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNBR, four pumps are required at rated power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

BASES

APPLICABILITY (continued)

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) Loops - High Water Level"); and
LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

ACTIONS

A.1 and A.2

If the requirements of the LCO are not met, the first Required Action is to reduce power to a level acceptable for short term three loop operation. This lower power level reduces the core heat removal needs and minimizes the possibility of violating DNB limits. The reduced power level allows for a restart of the RCP without violating safety analysis limits.

The second Required Action is to restore the RCS loop to operation.

The Completion Times of 15 minutes to reduce power is reasonable considering a Partial Reactor Trip, which is not credited in the accident analysis, will automatically perform this function. The 2 hours to restore the loop to operation is reasonable to perform minor repairs and prestart checkouts of an RCP removed from service.

B.1

If the Required Action and associated Completion Time of Condition A is not met or the requirements of the LCO are not met for other reasons the Required Action is to bring the plant to MODE 3. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours that each RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB limit. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

BASES

REFERENCES	1. Chapter 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

APPLICABLE SAFETY ANALYSES

Whenever the Control Rod Drive Control System (CRDCS) is capable of rod withdrawal an inadvertent rod withdrawal from subcritical is possible, resulting in a power excursion. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible irregardless of CRDCS capabilities. Such a transient could be caused by the mechanical failure of a CRDM.

Therefore, in MODE 3 with the CRDCS capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires four RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the CRDCS is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS Loops - MODE 3 satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The purpose of this LCO is to require that RCS loops be OPERABLE to support the specified conditions. In MODE 3 with the CRDCS capable of rod withdrawal, four RCS loops must be OPERABLE and in operation. Four RCS loops are required to be in operation in MODE 3 with the CRDCS capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

When the CRDCS is not capable of rod withdrawal, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that sufficient redundancy exists.

The Note permits all RCPs to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.

The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the stopping of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

BASES

LCO (continued)

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

c. The CRDCS is not capable of rod withdrawal.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, four RCS loops OPERABLE and four RCS loops in operation, applies to MODE 3 with the CRDCS capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the CRDCS not capable of rod withdrawal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4, "Residual Heat Removal (RHR) Loops - High Water Level"; and
LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

BASES

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for decay heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without a redundant loop because of the low probability of a failure in the remaining loops occurring during this period.

B.1

If restoration for Required Action A.1 is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1

If one required RCS loop is not in operation, and the CRDCS is capable of rod withdrawal, the Required Action is to place the CRDCS in a condition incapable of rod withdrawal (e.g. removing power from CRDMs). When the CRDCS is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of four RCS loops in operation. If three RCS loops are in operation, the CRDCS must be rendered incapable of rod withdrawal. The Completion Time of 2 hours to defeat the CRDCS is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

D.1, D.2, and D.3

If two or more required RCS loops are inoperable, or two or more the required RCS loops are not in operation when the CRDCS is capable of rod withdrawal or one required RCS loop is not in operation when the CRDCS is not capable of rod withdrawal the CRDCS must be placed in a condition incapable of rod withdrawal (e.g. removing power to the CRDMs). All operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum

BASES

ACTIONS (continued)

SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and removing power from the CRDMs removes the possibility of an inadvertent rod withdrawal. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until the required are restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required loops are in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 20% for required RCS loops. If the SG secondary side narrow range water level is < 20%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

Verification that each required RCP is OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required RCP.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES Chapter 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND	In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the Low Head Safety Injection (LHSI) heat exchangers connected to the Residual Heat Removal (RHR) System. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid. The reactor coolant is circulated through four RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification. In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or two RHR loops for decay heat removal and transport. The flow provided by one RCP loop or two RHR loops is adequate for decay heat removal. The other intent of this LCO is to require that additional paths be available to provide redundancy for decay heat removal.
APPLICABLE SAFETY ANALYSES	In MODE 4, RCS loop circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation. RCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(d)(2)(ii).
LCO	The purpose of this LCO is to require that at least two RCS loops or three RHR loops are OPERABLE in MODE 4, and that one of the RCS loops or two of the RHR loops are in operation. Any one RCS loop or two RHR loops in operation will provide enough flow to remove the decay heat from the core and allow for cooldown. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

BASES

LCO (continued)

The Note permits all RCPs or LHSI pumps to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the Note is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" therefore maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG, which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE LHSI pump capable of providing forced flow to an OPERABLE LHSI heat exchanger. RCPs and LHSI pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

BASES

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RCS or two loops of RHR provides sufficient circulation for these purposes. However, additional loops consisting of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) Loops - High Water Level"; and
LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

ACTIONS

A.1 and A.2

If one required loop is inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If two or more required loops are inoperable or the required loop(s) are not in operation, except during conditions permitted by the Note in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until the required loop(s) are restored to OPERABLE status or operation.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that the required RCS or RHR loops are in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 20%. If the SG secondary side narrow range water level is < 20%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

Verification that each required pump is OPERABLE ensures that an additional RCS or LHSI / RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation. This is acceptable because the proper breaker alignment and power availability are ensured if a pump is operating.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of four RHR loops connected to the RCS, each loop containing an LHSI heat exchanger, an LHSI pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One LHSI pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for mixing and decay heat removal. The other intent of this LCO is to require that additional paths be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be one RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels ≥ 20 % to provide an alternate method for decay heat removal via natural circulation (Ref. 1).

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops - MODE 5, Loops Filled satisfies Criterion 4 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level ≥ 20%. One RHR loop is required for forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels ≥ 20%. Should the operating RHR loops fail, the SGs could be used to remove the decay heat via natural circulation.

Note 1 permits all LHSI pumps to be removed from operation ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met:

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" therefore maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

BASES

LCO (continued)

Note 2 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

LHSI pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ≥ 20%.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2;"
LCO 3.4.5, "RCS Loops - MODE 3;"
LCO 3.4.6, "RCS Loops - MODE 4;"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled;"
LCO 3.9.4, "Residual Heat Removal (RHR) Loops - High Water Level"; and;
LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

ACTIONS

A.1, A.2, B.1, and B.2

If one RHR loop is OPERABLE and either the required SGs have secondary side water levels < 20%, or one required RHR loop is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

BASES

ACTIONS (continued)

C.1 and C.2

If a required RHR loop is not in operation, except during conditions permitted by Note 1, or if no required loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are ≥ 20% ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.3

Verification that each required RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required RHR pump. If secondary side water level is ≥ 20% in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation. This is acceptable because proper breaker alignment and power availability are ensured if a pump is operating.

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND	In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid. In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.
APPLICABLE SAFETY ANALYSES	In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing. RCS loops in MODE 5 (Loops Not Filled) satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).
LCO	The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running LHSI pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations. An OPERABLE RHR loop is comprised of an OPERABLE LHSI pump capable of providing forced flow to an OPERABLE LHSI heat exchanger. LHSI pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

BASES

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2;"
LCO 3.4.5, "RCS Loops - MODE 3;"
LCO 3.4.6, "RCS Loops - MODE 4;"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled;"
LCO 3.9.4, "Residual Heat Removal (RHR) Loops - High Water Level;" and
LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

ACTIONS

A.1

If one required RHR loop is inoperable, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required loop is OPERABLE or the required loop is not in operation, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification every 12 hours that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters and their controls, and Chemical Volume and Control System (CVCS) valves that limit the increase in water level. Pressurizer Safety Relief Valves are addressed by LCO 3.4.10, "Pressurizer Safety Relief Valves (PSRVs)."

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of emergency supply pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

BASES

BACKGROUND (continued)

On high pressurizer level, actions are required to avoid filling of the pressurizer, which would lead to pressurization and opening of the PSV with water overflow. The CVCS charging and auxiliary spray valves isolate on an increasing pressurizer level to perform this function. The CVCS charging isolation is normally open during operation for pressurizer level control while the CVCS auxiliary spray valve is opened during plant cooldown to reduce pressurizer temperature and provide pressure control when the RCS normal sprays are not functional (RCPs removed from service).

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in Chapter 15 (Ref. 1) do not take credit for emergency supply pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

On high pressurizer level, actions are required to avoid filling of the pressurizer, which would lead to pressurization and opening of the PSRV with water overflow. The closure of the CVCS charging and auxiliary spray valves perform this function.

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii). Although the emergency supply heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1240 cubic feet, which is equivalent to 75%, ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

BASES

LCO (continued)

The LCO requires three groups of OPERABLE emergency supply pressurizer heaters, each with a capacity ≥ Q/2 kW. Four groups of pressurizer heaters are provided on separate emergency buses. Two half-capacity groups of heaters are assumed available assuming a single failure. The minimum heater capacity provided by two heater groups is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of Q kW is derived from the use of 12 heaters rated at 24 kW each. The amount needed to maintain pressure is dependent on the heat losses.

The LCO requires the CVCS charging and auxiliary spray valves to be OPERABLE to prevent overfilling the pressurizer which would lead to pressurization and opening of the PSRV with water overflow. For the valves to be OPERABLE they must be capable of automatically closing on the CVCS Charging Isolation on Pressurizer Level signal generated from the Protection System (LCO 3.3.1, "Protection System (PS).")

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is a need to maintain the availability of the emergency supply pressurizer heaters. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1, A.2, A.3, and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Level > Max 1p Trip Setpoint.

BASES

ACTIONS (continued)

If the pressurizer water level is not within the limit, action must be taken to bring the plant to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3 with all rods fully inserted and incapable of withdrawal. Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1

If one required group of emergency supply pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using the remaining heaters.

C.1

If the CVCS charging valve or the CVCS auxiliary spray valve is inoperable and cannot be restored in the allowed Completion Time the associated flow path must be isolated. This ensures the function of the valves has been performed. The flow path can be isolated using additional valves which are in the flow path, but do not get the automatic closure signal on increasing pressurizer level. The Completion Time is reasonable considering the controls for the additional valves are located in the control room.

D.1 and D.2

If one required group of emergency supply pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumption of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated emergency supply pressurizer heaters are verified to be at their design rating. This SR may be verified by energizing the heaters and measuring circuit current. The frequency of 92 days is considered adequate to detect heater degradation and has shown by operating experience to be acceptable.

SR 3.4.9.3

These Surveillances demonstrate that each automatic valve used to isolate the pressurizer actuates to the required position on an actual or simulated PS signal. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of Protection System testing, and equipment performance is monitored as part of the Inservice Testing Program.

REFERENCES

1. Chapter 15.

2. NUREG-0737, November 1980.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Relief Valves (PSRVs)

BASES

BACKGROUND

The Pressurizer Safety Relief Valves (PSRVs), in conjunction with the Reactor Protection System (RPS), provide overpressure protection for the RCS. The PSRVs are pilot operated relief valves. Each relief valve has two types of pilot valves. The spring operated pilot valves are used to detect and provide the opening setpoint for the relief valve. The second type of pilot valve is the solenoid type. The solenoid pilot valves are used only for low temperature overpressure protection.

Each PSRV has two spring operated pilot valves that are arranged in a parallel configuration; only one pilot valve has to be operated to allow the PSRV to open. Both pilot valves are tested and adjusted to the proper relief valve setpoint. Normally one pilot valve is in operation (un-isolated) and the other is manually isolated and considered a standby device. In the event the in service pilot valve is suspected of being faulty the standby pilot can be placed in service, and the other valve isolated for troubleshooting. If it is determined that both pilot valves are faulty the PSRV would be declared inoperable. The PSRVs are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2803 psia, which is 110% of the design pressure.

Because each of the three safety relief valves are totally enclosed and self actuating they are considered independent components. The relief capacity for each valve, 661,400 lb_m/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the PSRVs is directed to the Pressurizer Relief Tank. This discharge flow is indicated by an increase in temperature downstream of the PSRVs or increase in the Pressurizer Relief Tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures less than or equal to LTOP arming temperature specified in the PTLR, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by meeting the requirements of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

BASES

BACKGROUND (continued)

The PSRV lift settings include a ± 2% tolerance requirement for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established. The set point tolerance for intermediate shutdown conditions is ± 45 psi.

The PSRVs are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses (Ref. 2) that require safety valve actuation assume operation of three PSRVs to limit increases in RCS pressure. Detailed analyses of the transients is contained in Reference 2. Compliance with this LCO is consistent with the design bases and accident analyses assumptions. The overpressure protection analysis (Ref. 3) is also based on operation of three PSRVs.

Pressurizer Safety Relief Valves satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The three PSRVs are set to open at ≥ 2484.3 and ≤ 2585.7 psig to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The lift settings include a ± 2% tolerance requirement for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety relief valves for protection.

BASES

APPLICABILITY (continued)

The LCO is not applicable in MODE 4 when any RCS cold leg temperatures are less than or equal to the LTOP arming temperature specified in the PTLR. or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head removed.

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the PSRVs at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one PSRV inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the Pressurizer Relief System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If Required Action cannot be met within the required Completion Time or if two or more PSRVs are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures less than or equal to the LTOP arming temperature specified in the PTLR within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures less than or equal to the LTOP arming temperature specified in the PTLR overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large Pressurizer insurges, and thereby removes the need for overpressure protection by three PSRVs.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.10.1 SRs are specified in the Inservice Testing Program. PSRVs are to be tested in accordance with the requirements the ASME Code (Ref. 3), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified. The PSRV setpoint is ± 2 % for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.
REFERENCES	1. ASME, Boiler and Pressure Vessel Code, Section III. 2. Chapter 15. 3. ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants, 2004.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Low Temperature Overpressure Protection (LTOP)

BASES

BACKGROUND

LTOP controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the Pressurizer Safety Relief Valves (PSRVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all medium head safety injection (MHSI) pumps have their respective miniflow lines open and isolating the cold leg accumulators. The pressure relief capacity requires either two redundant PSRVs which are LTOP capable or a depressurized RCS and an RCS vent of sufficient size. Two PSRVs or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core

BASES

BACKGROUND (continued)

decay heat levels, the makeup system can provide adequate flow via the makeup control valve. By having the MHSI miniflow lines open, the MHSI pumps can be made available in the event of loss of inventory.

The RCPS are administratively restricted from being started unless the secondary side water temperature of each steam generator is less than or equal to 50°F above each of the RCS cold leg temperatures.

The LTOP System for pressure relief consists of two PSRVs with reduced lift settings, or a depressurized RCS and an RCS vent of sufficient size. One PSRV has adequate relieving capability to keep from overpressurization for the assumed coolant input capability. Two PSRVs are required for redundancy.

PSRV Requirements for LTOP Capability

As designed for the LTOP System, each PSRV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic in Technical Specification Section 3.3. The LTOP actuation logic monitors the RCS pressure and determines when a condition not acceptable in the PTLR limits is approached. If the indicated pressure meets or exceeds the calculated value, a PSRV is signaled to open.

The PTLR presents the PSRV setpoints for LTOP. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of the valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PSRV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PSRV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

In addition to the LTOP actuation logic the PSRVs are considered LTOP capable if they have the appropriate power supplies to perform their overpressure protection function.

BASES

BACKGROUND (continued)

RCS Vent Requirements for LTOP Capability

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement (Ref.1) , it requires an open, clear flowpath and disabling any valves in the open position which could potentially block the flowpath. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding the LTOP arming temperature specified in the PTLR, the PSRVs will prevent RCS pressure from exceeding the Reference 1 limits. At about the LTOP arming temperature specified in the PTLR, and below, overpressure prevention falls to two OPERABLE PSRVs or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the PSRV setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using PSRV method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

BASES

APPLICABLE SAFETY ANALYSES (continued)

Mass Input Type Transients

a. Safety Injection; or

b. Charging / letdown flow mismatch.

Heat Input Type Transients

Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Ensuring the MHSI miniflow lines are open on MHSI pumps capable of injecting into the RCS;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Disallowing start of an RCP if secondary temperature is more than 50 °F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection.

The Reference 4 analyses demonstrate that either one PSRV or the depressurized RCS and RCS vent can maintain RCS pressure below limits when the MHSI miniflow lines are open and the pumps are actuated. Thus, the LCO allows MHSI pumps to be available during the LTOP MODES. Since neither the PSRV nor the RCS vent can handle the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulators isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of accumulator discharge is over a narrower RCS temperature range (175°F and below) than that of the LCO (248°F and below).

The RCPS are administratively restricted from being started unless the secondary side water temperature of each steam generator is less than or equal to 50°F above each of the RCS cold leg temperatures.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fracture mechanics analyses established the temperature of LTOP Applicability at the LTOP arming temperature specified in the PTLR.

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6), requirements by having a minimum of three MHSI pumps and SI actuation enabled.

PSRV Performance

The fracture mechanics analyses show that the vessel is protected when the PSRVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of four MHSI pumps at various temperatures from 70°F to 250°F injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PSRV opening and closing, resulting from signal processing and valve stroke times. The PSRV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

The PSRV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PSRVs solenoid valves are considered active components. Thus, the failure of one PSRV is assumed to represent the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 10.1 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, all MHSI pumps available, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

This LCO requires that LTOP is OPERABLE. LTOP is OPERABLE when the minimum coolant mass and heat input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant mass and heat input capability, the LCO requires that all accumulator discharge isolation valves be closed, the miniflow line be open for each MHSI pump capable of injecting into the RCS, and restrictions be placed on the starting of an RCP.

The LCO is modified by a Note. This Note states that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

The elements of the LCO that provide pressure relief capabilities are:

a. Two OPERABLE PSRVs,

A PSRV is OPERABLE for LTOP when its lift setpoint is set to the limit required by the PTLR, testing proves its ability to open at this setpoint, and it is powered from an essential source.

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of ≥ 10.1 square inches.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP arming temperature specified in the PTLR, in MODE 5, and in MODE 6 when the reactor vessel head is on. The PSRVs provide overpressure protection that meets the Reference 1 P/T limits above the LTOP arming temperature specified in the PTLR. When the reactor vessel head is off, overpressurization cannot occur.

BASES

APPLICABILITY (continued)

LCO 3.4.3, "RCS P/T Limits" provides the operational P/T limits for all MODES. LCO 3.4.10, "PSRVs," requires the OPERABILITY of the PSRVs that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above the LTOP arming temperature specified in the PTLR.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

A Note prohibits the application of LCO 3.0.4 b to an inoperable LTOP System. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1 and A.2

With the miniflow line of any MHSI pump not open and the MHSI pump capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

B.1, C.1, and C.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to greater than the LTOP arming temperature specified in the PTLR, an accumulator pressure of 450 psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

BASES

ACTIONS (continued)

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

D.1

In MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP arming temperature specified in the PTLR, with one required PSRV valve inoperable, the PSRV must be restored to OPERABLE status within a Completion Time of 72 hours. Two PSRVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the PSRVs are required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

E.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two PSRVs inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 12 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.

F.1

The RCS must be depressurized and a vent must be established within 6 hours when:

a. Two or more required PSRVs are inoperable;

b. Any Required Action and associated Completion Time of Condition D or E is not met; or

c. The LTOP System is inoperable for reasons other than Condition A, B, D, or E.

BASES

ACTIONS (continued)

The vent must be sized ≥ 10.1 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

To minimize the potential for a low temperature overpressure event by limiting the mass input capability the accumulator discharge isolation valves are verified closed and locked out.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.11.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability the miniflow lines for each MHSI capable of injecting into the RCS is verified open. If the miniflow lines are not open an alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control circuit being disabled and at least one valve in the discharge flow path being closed.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.3

The RCS vent of ≥ 10.1 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked (valves that are sealed or secured in the open position are considered "locked" in this context) or

b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position). Two removed PSRVs, a train of Primary Depressurization System (PDS) valves open and de-energized, or an open Pressurizer manway also fits this category.

The passive vent path arrangement must only be open to be OPERABLE. This Surveillance is required to be met if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.11 b.

SR 3.4.11.4

To minimize the potential for a low temperature overpressure event by limiting the heat input capability restrictions are placed on the starting of an RCP. Verification that the temperature of the secondary side water is within the limits assumed in the overpressure protection analysis ensures a heat input overpressure event will not occur.

Performing this surveillance within 15 minutes prior to the start of an RCP is sufficient, considering the indications available to the operator in the control room, to verify the required status of the equipment.

BASES

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. Chapter 15.
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.
8. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves that isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leak tight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE is 0.125 gallon per minute (gpm) per steam generator (SG) or increases to 0.125 gpm per SG as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through each SG to less than or equal to 150 gallons per day is lower than the primary to secondary leakage value used in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from variety of accidents (such as a main steam line break, steam generator tube rupture, rod ejection accident, RCS pump locked rotor, etc. The basic radiological acceptance criteria associated with the alternative source term (AST) methodology are found in 10 CFR 50.34(a)(1) for the offsite receptors, with a limit of 25 rem total effective dose equivalent (TEDE). 10 CFR Part 50, Appendix A, GDC 19 as incorporated by reference in 10 CFR 52.47(a)(1), includes the criteria for control room personnel (5 rem TEDE). These criteria, however, are used for evaluating potential reactor accidents of exceedingly low occurrence probability and low risk of public exposure to radiation. For events with higher probability of occurrence, the acceptance criteria for the offsite receptors are more stringent, while the criteria for the control room operators remains the same.

The RCS Operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE Through Each SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

BASES

APPLICABILITY (continued)

LCO 3.4.13, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, Pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, Pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.14, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

<u>SR 3.4.12.2</u>

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through each SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.16, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through each SG. If

BASES

SURVEILLANCE REQUIREMENTS (continued)

it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG, in which case the LEAKAGE rate limit of 150 gallons per day would still apply.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, Pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45 May 1973.
3. Chapter 15.
4. NEI 97-06, Steam Generator Program Guidelines.
5. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.12, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.12.1). A known component of the identified LEAKAGE before operation begins is the least of two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

The main purpose of this LCO is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident that could degrade the ability for low pressure injection.

PIVs are provided to isolate the RCS from the following connected systems:

a. Residual Heat Removal (RHR) System:

b. Safety Injection System: and

c. Chemical and Volume Control System.

BASES

BACKGROUND (continued)

The PIVs are listed in Reference 6.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Intersystem loss of coolant accidents can result from a postulated failure of the PIVs, which are part of the RCPB. Intersystem LOCAs result in a pressurization of the systems downstream of the PIVs from the RCS. The low pressure portions of the connecting systems are designed for moderate pressures, therefore overpressurization failure of the low pressure piping would result in a LOCA outside containment.

RCS PIV Leakage satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum of 5 gpm at an RCS pressure of ≥ 2215 psig and ≤ 2255 psig.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the residual heat removal flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the residual heat removal mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

BASES

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system OPERABILITY, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1, A.2.1, and A.2.2

If leakage from one or more RCS PIVs is not within limits the flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2.1 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation. Action A.2.2 provides for restoration of the leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

B.1 and B.2

If any required Action and associated completion Time can not be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable to reach the required plant conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2.1 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 24 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to

BASES

SURVEILLENCE REQUIREMENTS (continued)

be performed on valves in the residual heat removal flowpath when the RHR System is aligned to the RCS in the decay heat removal mode of operation. PIVs contained in the residual heat removal flow path must be leakage rate tested after the RHR loops are secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. Table 3.9.6-2.
7. ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants, 2004.
8. 10 CFR 50.55a(g).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE and air cooler condensate flow rate monitor are instrumented to alarm for increases of 0.5 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. The instrument sensitivity of 10^{-9} μCi/cc radioactivity for particulate monitoring is practical for this leakage detection systems.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew point is well within the sensitivity range of available instruments.

BASES

BACKGROUND (continued)

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump and condensate flow from air coolers. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in Reference 3. Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(d)(2)(ii).

LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

BASES

LCO (continued)

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor, in combination with a particulate radioactivity monitor, and a containment air cooler condensate flow rate monitor provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

With the required containment sump monitor inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.12.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.12.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, Pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required sump monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

BASES

ACTIONS (continued)

B.1.1, B.1.2, B.2.1, and B.2.2

With the particulate containment atmosphere radioactivity monitoring instrumentation channel inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.12.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere radioactivity monitor. Alternatively, continued operation is allowed if the air cooler condensate flow rate monitoring system is OPERABLE, provided grab samples are taken or water inventory balances performed every 24 hours.

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.12.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

C.1 and C.2

With the required containment air cooler condensate flow rate monitor inoperable, alternative action is again required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.12.1, must be performed to provide alternate periodic information. Provided the isotopic analysis or a water inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment air cooler condensate flow rate monitor to OPERABLE status.

The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE. A Note is added allowing that SR 3.4.12.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

BASES

ACTIONS (continued)

D.1 and D.2

With the required containment atmosphere radioactivity monitor and the required containment air cooler condensate flow rate monitor inoperable, the only means of detecting leakage is the containment sump monitor. This Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

E.1 and E.2

If a Required Action of Condition A, B, C, or D cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

A CHANNEL CHECK of the containment atmosphere radioactivity monitor provides reasonable confidence that the instrument is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.2, SR 3.4.14.3, and SR 3.4.14.4

These SRs require the performance of a CALIBRATION for each of the RCS leakage detection instrumentation channels. The CALIBRATION verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45.
3. Chapter 5.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Specific Activity

BASES

BACKGROUND

The maximum Total Effective dose equivalent (TEDE) an individual at the Exclusion Area Boundary can receive for 2 hours following an accident, or at the Low Population Zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1). Doses to the Control Room operators must be limited per GDC 19. The limits on specific activity ensure that the doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam generator tube rupture (SGTR) or a main steam line break (MSLB) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and Control Room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting offsite and Control Room doses meet the appropriate Standard Review Plan acceptance criteria following a SGTR or a MSLB accident. The SGTR safety analysis (Ref. 3) assumes the specific activity of the reactor coolant at, or more conservative than, the LCO limit. The MSLB safety analysis (Ref. 4) assumes the specific activity of the reactor coolant at, or more conservative than, the LCO limit and an existing reactor coolant system generator (SG) tube leakage rate of 0.125 gpm in the affected steam generator and 0.375 gpm combined in the unaffected steam generators. The safety analysis for both accidents assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.17, "Secondary Specific Activity."

The analysis for the MSLB and SGTR accidents establish the acceptance to limits for RCS specific activity.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Each of the above analyses must consider two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases, by a factor of 500, the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a MSLB or SGTR, respectively. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas specific activity is assumed to be 210 μCi/gm DOSE EQUIVALENT XE-133.

These analyses also assume a loss of offsite power at the same time as the SGTR or the MSLB event. The SGTR requires operator action to initiate a manual reactor trip after 30 minutes and the loss of offsite power is assumed at this time. The MSLB causes a reduction in reactor coolant temperature and pressure. The temperature decrease causes an increase in reactor power. A reactor trip is initiated on either a low SG pressure or high SG pressure decrease.

For the SGTR and the MSLB radiological analysis, the coincident loss of offsite power causes the turbine bypass valves to close to protect the condenser. For the SGTR, a rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the main steam relief valves. A failure to close of the main steam relief valve on the affected SG is also assumed. The unaffected SGs remove core decay heat by venting steam to the atmosphere until RHR system is placed in service. For the MSLB, an uncontrolled (i.e., released to atmosphere) blowdown of only one steam generator is assumed. The unaffected SGs remove core decay heat by venting steam to the atmosphere until RHR system is placed in service. Radioactively contaminated steam is released to the atmosphere through the faulted SG as well as the intact SGs assuming the primary to secondary leak rates shown above.

The applicable safety analysis shows the radiological consequences of either an SGTR or MSLB accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 60 μCi/gm DOSE EQUIVALENT I-131 for more than 48 hours.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO	The iodine specific activity in the reactor coolant is limited to 0.45 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 210 μCi/gm DOSE EQUIVALENT XE-133, as contained in SR 3.4.15.2 and SR 3.4.15.1 respectively. The limits on specific activity ensure that offsite and Control Room doses will meet the appropriate Standard Review Plan acceptance criteria (Ref. 2). The SGTR accident analysis (Ref. 3) and the MSLB accident analysis (Ref. 4) show that the calculated dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a MSLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).
APPLICABILITY	In MODES 1, 2, 3 and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENTXE-133 is necessary to limit the potential consequences of an SGTR and an MSLB to within the SRP acceptance criteria (Ref. 2). In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, monitoring of RCS specific activity is not required.
ACTIONS	<u>A.1 and A.2</u> With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to verify that the specific activity is ≤ 60.0 μCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend. The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were no iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a MSLB or SGTR occurring during this time period. A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

BASES

ACTIONS (continued)

B.1

With the DOSE EQUIVALENT XE-133 in excess of the allowed limit, DOSE EQUIVALENT XE-133 must be restored to within limits within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a MSLB or SGTR occurring during this time period.

A NOTE permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

C.1 and C.2

If any Required Action and the associated Completion Time of Condition A or B is not met or if the DOSE EQUIVALENT I-131 is > 60.0 μCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in noble gas specific activity.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

SR 3.4.15.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

REFERENCES

1. 10 CFR 50.67.
2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms."
3. Section 15.0.3.7.
4. Section 15.0.3.6.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops – MODES 1 and 2," LOC 3.4.5, "RCS Loops – MODE 3," LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.8, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.8, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.8. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES	The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.12, "RCS Operational LEAKAGE." The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via main steam relief trains and safety valves and the majority is discharged to the main condenser. The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.5 gallon per minute or is assumed to increase to 0.5 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.15, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits.) Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).
LCO	The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the Steam Generator Program. During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity. In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube. A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 0.125 gpm per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.12, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through each SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The Actions are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

<u>A.1 and A.2</u>

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.16.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination

BASES

ACTIONS (continued)

is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Time of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.16.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 100.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 RCS Loops - Test Exceptions

BASES

BACKGROUND

This special test exception to LCO 3.4.4, "RCS Loops - MODES 1 and 2," permits reactor criticality under no flow conditions during certain PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) to be performed while at low THERMAL POWER levels. Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant as specified in GDC 1, "Quality Standards and Records" (Ref. 2).

The key objectives of a test program are to provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the unit has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations.

The tests will include verifying the ability to establish and maintain natural circulation following a plant trip between 10% and 20% RTP, performing natural circulation cooldown on emergency power, and during the cooldown, showing that adequate boron mixture occurs and that pressure can be controlled using auxiliary spray and pressurizer heaters powered from the emergency power sources.

APPLICABLE SAFETY ANALYSES

The tests described above require operating the plant without forced convection flow and as such are not bounded by any safety analyses. However, operating experience has demonstrated this exception to be safe under the present applicability.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(d)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

BASES

LCO

This LCO provides an exemption to the requirements of LCO 3.4.4.

The LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without the LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODES 1 and 2), and the appropriate tests could not be performed.

In MODE 2, where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER is ≤ 5%.

The exemption is allowed even though there are no bounding safety analyses. However, these tests are performed under close supervision during the test program and provide valuable information on the plant's capability to cool down without offsite power available to the reactor coolant pumps.

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS without any forced RCS flow. The LCO only allows testing under these conditions while in MODE 2. This testing is performed to establish that heat input from nuclear heat does not exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.

ACTIONS

A.1

If THERMAL POWER increases to > 5% RTP, the reactor trip breakers must be opened immediately in accordance with Required Action A.1 to prevent operation of the fuel beyond design limits.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.17.1 THERMAL POWER must be verified to be within limits once per hour to ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Plant operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.
REFERENCES	1. 10 CFR 50, Appendix B, Section XI. 2. 10 CFR 50, Appendix A, GDC 1, 1988.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when flow from the accumulators or safety injection (SI) begins (Ref. 1).

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of SI water.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. Boric acid used in the accumulators is enriched in B^{10} to allow for a reduction in the boric acid concentration. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

BASES

APPLICABLE SAFETY ANALYSES	The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large break LOCA is a double ended guillotine break in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. The Protection System automatically starts the Medium Head Safety Injection (MHSI) and Low Head Safety Injection (LHSI) pumps and initiates a partial cooldown of the secondary system. The degree of accumulator discharge into the RCS depends on RCS pressure.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase in water volume is a peak clad temperature penalty. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of 1236 ft^3 and 1412.6 ft^3.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 3).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 2000 psig, and the limits established in the SRs for contained volume, boron concentration, boron isotopic inventory, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. During RCS cooldown, two accumulators (Trains 3 and 4) are depressurized to approximately 304 psig and reconnected to the RCS to prevent Reactor Coolant Pump (RCP) seal injection damage in the event of an inadvertent RCS depressurization when the pressurizer is in a water solid state. Once all RCPs are stopped, the Train 3 and 4 accumulators are again isolated.

BASES

ACTIONS

A.1

If the boron concentration or boron enrichment of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration or enrichment limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, the main steam line break analysis demonstrates that the accumulators do not discharge following a large main steam line break. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration and enrichment to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration or enrichment, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and RCS pressure reduced to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator valve should be verified to be fully open every 12 hours. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling the affected accumulator within 6 hours after a 145 gallon (1%) volume increase will identify whether inleakage has caused a reduction in boron

BASES

SURVEILLANCE REQUIREMENTS (continued)

concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the in-containment refueling water storage tank (IRWST), because the water contained in the IRWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 4).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the RCS pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

SR 3.5.1.6

The boron used in the accumulators is enriched to > 37% in the B^{10} isotope. Verification every 24 months that the B^{10} enrichment is > 37% ensures that the B^{10} concentration assumed in the accident analysis is available. Since B^{10} in the accumulators is not exposed to a significant neutron field, 24 months is considered conservative.

REFERENCES

1. FSAR Chapter 15.
2. 10 CFR 50.46.
3. FSAR Chapter 6.
4. NUREG-1366, February 1990.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are two phases of ECCS operation: injection and hot leg recirculation. In the injection phase, water is taken from the in-containment refueling water storage tank (IRWST) and injected into the Reactor Coolant System (RCS) through the cold legs. After approximately 24 hours, the LHSI flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of two separate subsystems: Medium Head Safety Injection (MHSI) and Low Head Safety Injection (LHSI). Each subsystem consists of four redundant, 100% capacity trains. The ECCS accumulators and the IRWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

Each ECCS flow path consists of piping, valves, heat exchangers, and pumps such that water from the IRWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the MHSI pumps, the LHSI pumps, and heat exchangers. Each of the two subsystems (MHSI and LHSI) consists of four 100% capacity trains that are independent and redundant such that each train is capable of supplying 100% of the flow required to mitigate the accident consequences.

BASES

BACKGROUND (continued)

Four separate suction supply lines supply water from the IRWST to the ECCS pumps. Each of the four trains is independent and injects into a single RCS cold leg. If it is necessary to remove one LHSI train from service, an isolatable ECCS cross-connect ensures LHSI delivery in the event of a cold-leg break. Whenever the cross-connects are opened, the isolation valve’s electrical breakers are racked-out to avoid single failure. Otherwise, both ECCS cross-connects are isolated to maintain train separation.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the MHSI pumps, the secondary side is cooled down to approximately 870 psia at a rate of approximately 180°F/hr by means of the relief valves to ensure adequate injection from the MHSI system.

Due to the large miniflow lines, it is not necessary to limit the number of MHSI or LHSI pumps in service during low temperature conditions in the RCS. Refer to the Bases for LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for the basis of low RCS temperature operation.

The ECCS subsystems are actuated upon receipt of a Protection System (PS) signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the IRWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "In-Containment Refueling Water Storage Tank (IRWST)," provide the Cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The MHSI and LHSI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the MHSI pumps. The SGTR and MSLB events also credit the MHSI pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power; and

b. A small break LOCA event, with a loss of offsite power.

In the event of a large break LOCA, when the only available LHSI connection is located adjacent to the broken cold leg, ECCS delivery to the reactor vessel downcomer may be affected by steam entrainment to the broken leg. This assumes that one train is out of service due to preventative maintenance, one train is assumed to have a single failure, and another train feeds the broken loop. In order to mitigate the effect of degraded ECCS delivery due to steam entrainment, isolable ECCS cross-connects are provided. This arrangement directs a portion of the LHSI flow to an adjacent train, thereby reducing flow lost to steam entrainment. The ECCS cross-connects between Trains 1 and 2 and Trains 3 and 4 are normally isolated by two motor-operated valves in series to maintain train separation. Both cross-connect isolation valves are opened when an ECCS train is taken out of service for maintenance and power removed from the motor operators.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

APPLICABLE SAFETY ANALYSES (continued)

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boil off rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the MHSI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the MHSI pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling. The cooling capability of the steam generators is enhanced by the operation of the secondary side main steam relief trains.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Four 100% capacity independent (cross-connect closed) ECCS trains are required to ensure that sufficient ECCS flow is available. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

An ECCS train consists of an MHSI subsystem, and an LHSI subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of injecting upon an PS signal.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the IRWST to the RCS via the ECCS pumps to the individual cold leg injection nozzles. In the long term, this flow path may be switched to supply its flow to the RCS hot and cold legs.

The IRWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS pump operation.

To be considered OPERABLE, the IRWST must meet the water volume and boron concentration limits established in the SRs.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The MHSI pump performance requirements are based on a small break LOCA. MODE 2, and 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above. Below 356°F, the PS signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS - Shutdown."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "LHSI / RHR and Coolant Circulation - High Water Level," and LCO 3.9.5, " LHSI / RHR and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one MHSI train inoperable, the inoperable components must be returned to OPERABLE status within 120 days. The 120 day Completion Time is based on the assumption in the FSAR Chapter 15 analysis that one ECCS train is assumed out of service for maintenance at the time of the accident.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

B.1 and B.2

With one LHSI train inoperable, an acceptable ECCS configuration can be achieved by opening both ECCS cross connections. In the event of a cold leg break, one train is assumed lost due to steam entrainment to the broken loop, one train is assumed to mitigate the event, one train is assumed to spill out the break, and one train is assumed to have a single failure. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours. The 120 day Completion Time is based on the assumption in the FSAR Chapter 15 analysis that one ECCS train is assumed out of service for maintenance at the time of the accident.

BASES

ACTIONS (continued)

C.1

With two MHSI trains inoperable, at least one train must be restored to OPERABLE status in 72 hours. This allowed completion time is reasonable since two trains are available and only one train is required to accomplish the safety function. With only two trains OPERABLE, the single failure criterion is not met.

D.1 and D.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

Condition E is applicable with three or more trains inoperable. With less than 100% of the ECCS flow equivalent to two OPERABLE ECCS trains available, the facility is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verifying the correct alignment for manual, power operated and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. The ECCS flow path verification includes verification that the cold leg cross-connect valves are in their required position. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

The ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following a PS signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

SR 3.5.2.3

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.4 and SR 3.5.2.5

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated PS signal and that each ECCS pump starts on receipt of an actual or simulated PS signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of Protection System testing, and equipment performance is monitored as part of the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.6

Periodic inspections of the suction inlet from the IRWST ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR Section 6.2, "Containment Systems."
4. FSAR Chapter 15, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND	The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications. In MODE 4, a single ECCS train consisting of a Medium Head Safety Injection (MHSI) train is capable of providing the core cooling function. A second train is assumed to spill out of the break. Low head Safety Injection is not automatically actuated. The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the in-containment refueling water storage tank (IRWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.
APPLICABLE SAFETY ANALYSES	The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section. Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. Below P14 and RHR connected, LHSI is not automatically actuated by the Protection System (PS). However, MHSI is automatically actuated by the PS. Two trains of ECCS are required for MODE 4. Protection against single failures is not relied on for this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).
LCO	In MODE 4, two of the four independent (and redundant) ECCS MHSI trains are required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA. One train is required to accomplish the safety function and one train is assumed to feed the break. The ECCS cross-connects are not needed for events postulated in MODE 4. In MODE 4, an ECCS train consists of an MHSI subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the IRWST.

BASES

LCO (continued)

During an event requiring ECCS MHSI actuation, a flow path is required to provide an abundant supply of water from the IRWST to the RCS via the ECCS pumps and to its associated four cold leg injection nozzles. In the long term, this flow path may be switched to deliver its flow to the RCS hot and cold legs.

APPLICABILITY

In MODES 1, 2, 3 and 4, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4, two OPERABLE ECCS MHSI trains are acceptable without single failure consideration on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "LHSI/RHR and Coolant Circulation - High Water Level," and LCO 3.9.5, "LHSI/RHR and Coolant Circulation - Low Water Level."

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable ECCS MHSI train. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS MHSI train and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With one required MHSI train inoperable, the inoperable train must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

BASES

ACTIONS (continued)

B.1

When Required Action A.1 cannot be completed within the required Completion Time; or if two required ECCS MHSI trains are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 5 within 12 hours.

The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from MODE 4 in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

The applicable references from Bases 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 In-Containment Refueling Water Storage Tank (IRWST)

BASES

BACKGROUND

The IRWST supplies borated water to the refueling pool during refueling, and to the ECCS during accident conditions.

The IRWST supplies all four trains of the ECCS through separate, independent supply headers during the injection phase of a loss of coolant accident (LOCA) recovery.

During normal operation in MODES 1, 2, and 3, Medium Head Safety Injection (MHSI) and Low Head Safety Injection (LHSI) pumps are aligned to take suction from the IRWST.

The ECCS pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

This LCO ensures that:

a. The IRWST contains sufficient borated water to support the ECCS accident mitigation function; and

b. The reactor remains subcritical following a LOCA.

Insufficient water in the IRWST could result in insufficient cooling capacity and suction head for ECCS operation. Improper boron concentrations or enrichment could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the IRWST provides a source of borated water to the ECCS pumps. As such, it provides containment energy removal, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating," and B 3.5.3, "ECCS – Shutdown." These analyses are used to assess changes to the IRWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

The IRWST must also meet volume, boron concentration, boron isotopic inventory (i.e., enrichment), and temperature requirements for non-LOCA

BASES

APPLICABLE SAFETY ANALYSES (continued)

events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the required volumes for an outage and is therefore not limiting. The minimum IRWST volume is determined by ECCS pump NPSH requirements. The minimum boron concentration and isotopic inventory are explicit assumptions in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The importance of its value is small due to the Extra Boration System (EBS) with its high boron concentration.

The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the IRWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.

For a large break LOCA analysis, the minimum water volume of 500,342 gallons and the lower boron concentration limit of 1700 ppm of > 37% enriched boron are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core. This minimum volume bounds the ECCS pump NPSH requirements.

The maximum water volume of 523,703 gallons and the upper limit on boron concentration of 1900 ppm are used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

The upper temperature limit of 122°F is used in the small break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on IRWST water temperature are used to maximize the total energy release to containment.

The minimum temperature valve of 59°F is consistent with mechanical requirements, particularly reactor pressure vessel brittle fracture risk.

The IRWST satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO

The IRWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS pump operation.

To be considered OPERABLE, the IRWST must meet the water volume, and boron concentration and enrichment limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, IRWST OPERABILITY requirements are dictated by ECCS OPERABILITY requirements. Since the ECCS must be OPERABLE in MODES 1, 2, 3, and 4, the IRWST must also be OPERABLE to support its operation. In MODES 5 and 6, the IRWST is in standby. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "LHSI/RHR and Coolant Circulation - High Water Level," and LCO 3.9.5, "LHSI/RHR and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With IRWST boron concentration or enrichment not within limits, it must be returned to within limits within 8 hours. Under these conditions the ECCS cannot perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the IRWST boron concentration or enrichment to within limits was developed considering the time required to change the boron concentration/isotopic inventory and the fact that the contents of the tank are still available for injection.

B.1

With the IRWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, the ECCS cannot perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the IRWST is not required. The short time limit of 1 hour to restore the IRWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

BASES

ACTIONS (continued)

C.1 and C.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The IRWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

SR 3.5.4.2

The IRWST water volume should be verified every 7 days to be within limits. The required minimum volume is verified in order to ensure that a sufficient NPSH is available for injection and to support continued ECCS pump operation. The maximum volume is verified in order to ensure the value assumed in the post-LOCA boron precipitation evaluation is not exceeded. Since the IRWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

The boron concentration of the IRWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized.

Since the IRWST inventory is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.4

The boron used in the IRWST is enriched to > 37% in the B^{10} isotope. Verification every 24 months that the B^{10} enrichment is > 37% ensures that the B^{10} concentration assumed in the accident analysis is available. Since B^{10} in the IRWST is not exposed to a significant neutron field, 24 months is considered conservative.

REFERENCES

1. FSAR Chapter 6 and Chapter 15.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Extra Boration System (EBS)

BASES

BACKGROUND	The EBS is a manually actuated, safety-related system that is used in the mitigation of design basis accidents, including a steam generator tube rupture (SGTR). During this event, the EBS injects boron into the RCS to maintain the core subcritical while the RCS is being cooled to the point where the Low Heat Safety Injection System can be connected to remove core decay heat. The EBS also provides RCS makeup to balance a portion of the shrinkage during cooldown. The EBS can be used for hydrostatic testing of the RCS but otherwise does not perform any function supporting normal plant operation. The EBS consists of two identical trains. Each train is composed of its own boron tank, a high pressure 100% capacity pump, a test line, and injection lines to the RCS. The volume of concentrated boric acid required to maintain subcriticality is divided between the two EBS tanks. A common suction header allows either EBS pump to take suction from both tanks. The boron tanks and the primary train lines are filled with borated water and are located in a temperature controlled room to prevent crystallization of the boron (Ref. 1 and 2). Outside of the temperature controlled rooms, the EBS piping is filled with lower concentration borated water from the In-Containment Refueling Water Storage Tank.
APPLICABLE SAFETY ANALYSES	If needed, the EBS is manually initiated. A 30 minute operator action time is assumed in the analysis. Once started for safety reasons, the EBS will remain in operation until the boron concentration needed for cold shutdown is reached. The EBS is initiated for an SGTR to ensure adequate boration to prevent criticality. The contents of the EBS are not credited for core cooling or immediate boration in the LOCA analysis. The EBS maximum boron concentration of 7300 ppm is used in the Boron Precipitation Assessment (Ref. 2). The minimum boron concentration of 7000 ppm is credited in the SGTR analysis and for cooldown from other design basis events. Boron used in the EBS is enriched to ≥ 37% in the B^{10} isotope. The EBS minimum water volume limit of 2345 ft^3 total between the two EBS tanks is used to ensure that the appropriate quantity of highly borated water with sufficient negative reactivity is injected into the RCS to maintain the core in a shutdown condition following an SGTR or during cooldown for other Design Basis Accidents (DBAs). This volume includes approximately 175 ft^3 of unusable volume in each tank.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The minimum temperature limit of 68°F for the EBS borated water assures that the solution does not reach the point of boron crystallization.

The EBS satisfies Criteria 3 of 10 CFR 50.36(d)(2)(ii).

LCO

This LCO establishes the minimum requirements as well as requirements for contained volume, boron concentration, boron enrichment, and temperature of the EBS inventory (Ref. 3). This ensures that an adequate supply of borated water is available in the event of an SGTR or other design basis event to maintain the reactor subcritical following these accidents.

To be considered OPERABLE, the limits established in the SR for water volume, boron concentration, boron isotopic inventory, and temperature must be met.

APPLICABILITY

In MODES 1, 2, 3, and 4, the EBS is needed to maintain the core subcritical following an SGTR and during cooldown to MODE 5 for DBAs.

An SGTR and other DBAs that rely on the EBS for cooldown are not postulated in MODES 5, and 6 and EBS OPERABILITY is not required.

ACTIONS

A.1

If the boron concentration or boron enrichment of one or both EBS tanks is not within limits, it must be returned to within limits within 72 hours. Because of the low probability of an SGTR or other DBAs, the allowed Completion Time of 72 hours is acceptable and provides adequate time to restore concentration or enrichment to within limits.

B.1

If one EBS train is inoperable for reasons other than Condition A, the inoperable train must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE train is adequate to perform the shutdown function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE train could result in reduced EBS shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE train capable of performing the intended EBS function and the low probability of a DBA occurring.

BASES

ACTIONS (continued)

C.1

If both EBS trains are inoperable for reasons other than Condition A, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable, given the low probability of a DBA occurring.

D.1 and D.2

If any Required Action and associated Completion Time is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 3 within 12 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach MODE 5 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification every 24 hours that each EBS tank water temperature is at or above the specified minimum temperature is frequent enough to identify a temperature change that would approach the acceptable limit. The solution temperature is also monitored by an alarm that provides further assurance of protection against low temperature. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.5.2

Verification every 7 days that the EBS contained volume is above the required limit is frequent enough to assure that this volume will be available for quick injection into the RCS. If the volume is too low, the EBS would not provide enough borated water to ensure subcriticality during recirculation. Since the EBS volume is normally stable, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.5.3

Verification every 31 days that the boron concentration of each EBS tank is within the required limits ensures that the reactor remains subcritical following an SGTR or other DBA event and maintains the resulting IRWST pH in an acceptable range so that boron precipitation will not occur in the core. In addition, the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized.

SR 3.5.5.4

Verifying the correct alignment for manual and power operated valves in the EBS flow paths provides assurance that the proper flow paths will exist for EBS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.5.5

Demonstrating each EBS pump develops a flow rate ≥ 49.0 gpm and ≤ 55.4 gpm ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that the core will remain subcritical during and after a cooldown following design basis accidents including an SGTR. The maximum flow rate to the RCS is needed so that the pressurizer is not filled which could actuate the pressurizer relief valves. This inservice test confirms EBS pump OPERABILITY, trends performance, and detects incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.5.5.6

The boron used in each EBS tank is enriched to ≥ 37% in the B^{10} isotope. Verification every 24 months that the B^{10} enrichment is ≥ 37% ensures that the B^{10} concentration assumed in the accident analysis is available. Since B^{10} in the EBS is not exposed to a significant neutron field, 24 months is considered conservative.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.5.7

This Surveillance ensures that there is a functioning flow path from the EBS tank to the RCS. An acceptable method is to test the flow path in several separate tests. The 24 month Frequency is based on the need to perform this Surveillance under conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance test when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR Chapter 6

2. FSAR Chapter 15.

3. 10 CFR 50.46.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of a cylindrical reinforced concrete outer Shield Building, a cylindrical post-tensioned concrete inner Containment Building with a 0.25-inch thick steel liner, and an annular space between the two buildings. The containment, including all its penetrations, is a low leakage shell designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the containment provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

An annular space exists between the walls and domes of the Containment Building and the Shield Building to permit inservice inspection and collection of containment outleakage.

Containment piping penetration assemblies provide for the passage of process, service, sampling and instrumentation pipelines into the reactor vessel while maintaining containment OPERABILITY. The Shield Building allows controlled release of the annulus atmosphere under accident conditions, as well as protecting the Containment Building from external hazards.

The inner Containment Building and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

    1. Capable of being closed by an OPERABLE automatic containment isolation system; or

    2. Closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves."

BASES

BACKGROUND (continued)

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks;" and

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA, a main steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting Design Basis Accident. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25% per day in the safety analysis at P_a = 55 psig (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

BASES

LCO (continued)

Individual leakage rates specified for the containment air locks (LCO 3.6.2) and purge valves with resilient seals are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 La.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as containment post tensioning surveillance, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required Containment Leakage Rate Testing Program leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $1.0\ L_a$. At $1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Post Tensioning Surveillance Program. Testing and Frequency are in accordance with the ASME Code, Section III, Division 2, 2004 (Ref. 4).

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR Chapter 15.
3. FSAR Section 6.2.
4. ASME Code, Section III, Division 2, 2004.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND	Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation. One personnel air lock and one emergency air lock provide access to the Containment. Each air lock is nominally a right circular cylinder, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door). Each personnel air lock is provided with limit switches on both doors that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated. The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.
APPLICABLE SAFETY ANALYSES	The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 2). This leakage rate is defined in

BASES

APPLICABLE SAFETY ANALYSES (continued)

10 CFR 50, Appendix J, Option B (Ref. 1), as L_a = 0.25% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 55 psig following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during movement of recently irradiated fuel assemblies within containment are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel

BASES

ACTIONS (continued)

side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

BASES

ACTIONS (continued)

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

<u>B.1, B.2, and B.3</u>

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable.

BASES

ACTIONS (continued)

With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

BASES

ACTIONS (continued)

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment

BASES

SURVEILLANCE REQUIREMENTS (continued)

OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 months Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 months Frequency for the interlock is justified based on generic operating experience. The 24 months Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the airlock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR Section 6.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Isolation stage 1 (CI-1) occurs upon Containment pressure > MAX1p, or receipt of a Safety Injection System (SIS) signal. The CI-1 signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Isolation stage 2 (CI-2) occurs upon Containment pressure > MAX2p. The CI-2 signal isolates the remaining process lines, except systems required for accident mitigation.

As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

BASES

BACKGROUND (continued)

Full Flow Purge Sub-system (39 inch purge valves)

The Full Flow Purge portion of the Containment Ventilation System operates to supply outside air into the containment for ventilation and cooling or heating during a unit outage. The supply and exhaust lines each contain two isolation valves. The 39 inch full flow purge valves are qualified for automatic closure from their open position under DBA conditions. The Full Flow Purge Sub-system is not needed in MODES 1, 2, 3, and 4 and the 39 inch full flow purge valves are maintained closed to ensure the containment boundary is maintained.

Low Flow Purge Sub-system (20 inch purge valves)

The Low Flow Purge Sub-system operates to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access and

b. Equalize internal and external pressures.

Since the valves used in the Low Flow Purge Sub-system are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), fuel handling and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the 39 inch full flow purge valves are closed at the start of a LOCA or rod ejection but not for a fuel handling accident. The DBA analysis assumes that, within

BASES

APPLICABLE SAFETY ANALYSES (continued)

60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the design of the full flow purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves are pneumatically operated spring closed valves that will fail on the loss of air.

The full flow purge valves are designed to close in the environment following a LOCA or MSLB. However, the DBA dose analysis assumes that each full flow purge line is isolated during MODES 1, 2, 3, and 4.

The low flow purge valves may be opened during normal operation. In this case, the single failure criterion remains applicable to the low flow purge valves due to failure in the control circuit associated with each valve. The system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 39 inch full flow purge valves must be maintained sealed closed. The valves covered by this LCO are listed along with their associated stroke times in FSAR Section 6.2.4 (Ref. 2).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 2.

BASES

LCO (continued)

Purge valves with resilient seals must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during movement of recently irradiated fuel assemblies within containment are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for 39 inch full flow purge flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the full flow purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. A single full flow purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

BASES

ACTIONS (continued)

In the event the isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for purge valve leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This

Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

ACTIONS (continued)

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

B.1

With two or more containment isolation valves in one or more penetration flow paths inoperable, except for purge valve leakage not within limit, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

BASES

ACTIONS (continued)

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two or more containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Ref. 3. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices

BASES

ACTIONS (continued)

to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1, D.2, and D.3

In the event one or more full flow purge valves in one or more penetration flow paths are not within the full flow purge valve leakage limits, purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A full flow purge valve with resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.6. The specified Completion Time is reasonable, considering that one full flow purge valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification that those isolation devices outside containment capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the full flow purge valve with resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.3.6 must be performed at least once every 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the full flow purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.6, 184 days, is based on an NRC

ACTIONS (continued)

initiative, Generic Issue B-20 (Ref. 4). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

Required Action D.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

E.1 and E.2

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 39 inch full flow purge valve is required to be verified closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a full flow purge valve. The full flow purge valves are designed to close in the environment following a LOCA. However, the DBA dose analysis assumes that each full flow purge line is isolated during MODES 1, 2, 3, and 4.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2

This SR ensures that the low flow purge valves are closed as required or, if open, open for an allowable reason. If a low flow purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the low flow purge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The low flow purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.6

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 4).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES:

1. FSAR Chapter 15.

2. FSAR Section 6.2.

3. NUREG 0800, Section 6.2.4.

4. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."

5. Generic Issue B-24, "Seismic Qualification of Electrical and Mechanical Components."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND	The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or main steam line break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of transients which result in a negative pressure. Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values and may result in leakage greater than assumed in the accident analysis.
APPLICABLE SAFETY ANALYSES	Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and MSLB, which are analyzed using computer pressure transients. The worst case MSLB generates larger mass and energy release than the worst case LOCA. Thus, the MSLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1). The initial pressure condition used in the containment analysis was 15.96 psia (1.26 psig). This resulted in a maximum peak pressure from a LOCA of 52.0 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, results from the limiting LOCA. P_a is conservatively set at 55 psig. The maximum containment pressure resulting from the worst case LOCA, 52.0 psig, does not exceed the containment design pressure, 62 psig. The containment was also designed for an external pressure load equivalent to -3.0 psig An inadvertent actuation of the Severe Accident Heat Removal System is not considered a credible event for the U.S. EPR since it is manually actuated for beyond design basis events only. An evaluation of a Containment cooldown event determined a worse case pressure of -2.9 psig. Containment pressure satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO	Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following negative pressure transients.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.
ACTIONS	<u>A.1</u> When containment pressure is not within the limit of the LCO, it must be restored to within this limit within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour. <u>B.1 and B.2</u> If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.4.1 Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.
REFERENCES	1. FSAR Section 6.2. 2. 10 CFR 50, Appendix K.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND	The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or main steam line break (MSLB). The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Low Head Safety Injection System (LHSI) during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.
APPLICABLE SAFETY ANALYSES	Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1). The limiting DBAs considered relative to containment OPERABILITY are the LOCA and MSLB. The DBA LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to ESF Systems, assuming a worst case single active failure as identified in Reference 5.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting DBA for the maximum peak containment air temperature is an MSLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 131°F. This resulted in a maximum containment air temperature of 428°F. The containment has been qualified for this temperature (Ref. 2).

The temperature limit is used to establish the environmental qualification operating envelope for containment. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Refs. 3 and 4).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is an MSLB. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature and that required safety related equipment will continue to perform its function.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

BASES

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, a weighted average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. FSAR Section 6.2.

2. FSAR Section 3.8.

3. 10 CFR 50.49.

4. FSAR Section 3.11.

5. FSAR Section 15.0.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Shield Building

BASES

BACKGROUND	The shield building is a concrete structure that surrounds the Containment Building. Between the Containment Building and the shield building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the Containment Building. The Annulus Ventilation System (AVS) establishes a negative pressure in the annulus between the shield building and the containment building. Filters in the system then control the release of radioactive contaminants to the environment. The shield building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the AVS.
APPLICABLE SAFETY ANALYSES	The design basis for shield building OPERABILITY is a LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses. The shield building satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).
LCO	Shield building OPERABILITY must be maintained to ensure proper operation of the AVS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses. The LCO is modified by a Note allowing the shield building boundary to be opened intermittently under administrative controls. This Note only applies to openings in the shield building boundary that can be rapidly restored to the design conditions, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be accomplished by procedures, and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

BASES

APPLICABILITY	Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the containment following a LOCA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when a main steam line break, LOCA, or rod ejection accident could release radioactive material to the containment atmosphere. In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, shield building OPERABILITY is not required in MODE 5 or 6.
ACTIONS	<u>A.1</u> In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period. <u>B.1 and B.2</u> If the shield building cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	<u>SR 3.6.6.1</u> Verifying that shield building annulus negative pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Maintaining shield building OPERABILITY requires verifying each access opening door is closed. However, all shield building access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.6.6.3 and 3.6.6.4

The Annulus Ventilation System (AVS) exhausts the annulus atmosphere to the environment through appropriate treatment equipment. Each safety AVS train is designed to draw down the annulus to a negative pressure of ≥ 0.25 inches of water gauge (wg) in ≤ 305 seconds and maintain the annulus at a negative pressure ≥ 0.25 inches wg. To ensure that all fission products released to the annulus are treated, SR 3.6.6.3 and SR 3.6.6.4 verify that a pressure in the annulus that is less than the lowest postulated pressure external to the shield building boundary can be established and maintained. When the AVS System is operating as designed, the establishment and maintenance of annulus pressure cannot be accomplished if the shield building boundary is not intact. Establishment of this pressure is confirmed by SR 3.6.6.3, which demonstrates that the annulus can be drawn down to a negative pressure ≥ 0.25 inches wg using one AVS train. SR 3.6.6.4 demonstrates that the annulus can be maintained at a negative pressure ≥ 0.25 inches wg using one AVS train at a flow rate ≤ 1320 cfm. The primary purpose of these SRs is to ensure annulus boundary integrity. The secondary purpose of these SRs is to ensure that the AVS train being tested functions as designed. There is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the AVS System. These SRs need not be performed with each safety AVS train. The AVS train used for these Surveillances is staggered to ensure that in addition to the requirements of LCO 3.6.7, either safety AVS train will perform this test. The inoperability of the AVS System does not necessarily constitute a failure of these Surveillances relative to the shield building OPERABILITY. Operating experience has shown the shield building boundary usually passes these Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Annulus Ventilation System (AVS)

BASES

BACKGROUND

The AVS is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials that leak from the Containment Building into the shield building (secondary containment) following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The Containment Building is surrounded by a secondary containment called the shield building, which is a concrete structure. Between the Containment Building and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the Containment Building.

The AVS maintains a negative pressure in the annulus between the shield building and the Containment Building during operation. Filters in the system control the release of radioactive contaminants to the environment. Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the AVS. The AVS is designed to permit appropriate periodic pressure and functional testing to assure component integrity, OPERABILITY of active components, and operational performance of the system as required by GDC-43 "Testing of Containment Atmosphere Cleanup Systems" (Ref 4).

The AVS consists of one normal operation filtration train (non-safety related), and two independent and redundant accident filtration trains (safety related). The normal filtration train operates during normal plant operation, including cold shutdown and outages. During normal plant operation, the accident filtration trains are not required to be in operation, however they are both available for back-up if the normal filtration train is not able to maintain sufficient negative pressure in the annulus.

During normal operation, the conditioned air is drawn from the Nuclear Auxiliary Building Ventilation supply shaft to the bottom of annulus through a fire damper, manual regulated control damper, and two motor operated isolation dampers. The exhaust air is drawn through a vent at the top of annulus through two motor operated isolation dampers and fire dampers to the Nuclear Auxiliary Building Ventilation system exhaust fans via air shaft cell 3. See FSAR Section 9.4.3 (Ref. 5). The exhaust air from cell 3 is filtered by the pre-filter and HEPA filter and then discharged through the vent stack. The annulus air inlet and exhaust motor operated isolation dampers of the normal filtration train are the only components which are safety related.

BASES

BACKGROUND (continued)

The AVS accident filtration trains are used during a design basis event to contain leaks from the primary containment by maintaining a negative pressure in the annulus. During a design basis event, the annulus air is filtered before releasing to the environment. There are two independent 100% accident trains. Each train consists of an upstream air-tight motor controlled damper, electrical heater, pre-filter, upstream HEPA filter, an activated charcoal adsorber for removal of radio-iodines, downstream HEPA filter, downstream air-tight motor controlled damper, fan, and back-draft damper. The downstream bank of HEPA filters following the charcoal adsorber collects carbon particles and provide backup in case of failure of the main HEPA filter bank. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a containment isolation signal. The system is described in Reference 2.

The prefilters remove large particles in the air, and the moisture separators remove entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters may be included to reduce the relative humidity of the airstream on systems that operate in high humidity. Continuous operation of each train, for at least 10 hours per month, with heaters on, reduces moisture buildup on their HEPA filters and adsorbers.

During normal operation, the AVS normal operation filtration train (non-safety related) maintains a negative pressure in the annulus and processes the air through HEPA filters.

The AVS accident filtration train reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the AVS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The AVS design basis is to mitigate the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) assumes that only one train of the AVS is OPERABLE due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA. For all events analyzed, the AVS is assumed to be automatically initiated to reduce via filtration and adsorption, the radioactive material released to the environment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The modeled AVS actuation in the safety analyses is based upon a worst case response time following a containment isolation initiated at the limiting setpoint. The total response time, from exceeding the signal setpoint to attaining the negative pressure of 0.25 inches wg in the shield building, is 305 seconds. This response time is composed of signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting.

The AVS satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

In the event of a DBA, one AVS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two safety related trains of the AVS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the AVS is not required to be OPERABLE (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

ACTIONS

A.1

With one AVS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant AVS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

BASES

ACTIONS (continued)

B.1 and B.2

If the AVS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Operating each AVS train for ≥ 10 continuous hours ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating units indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.7.2

This SR verifies that the required AVS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.7.3

The automatic startup ensures that each AVS train responds properly. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the

BASES

SURVEILLANCE REQUIREMENTS (continued)

potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the AVS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.7.1.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. FSAR Section 6.2.
3. FSAR Chapter 15.
4. 10 CFR 50, Appendix A, GDC 43.
5. FSAR Section 9.4.3.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 pH Adjustment

BASES

BACKGROUND	The U.S. EPR design includes pH adjustment baskets that provide adjustment of the pH of the water in the containment following an accident where the containment floods. Following an accident with a large release of radioactivity, the containment pH is automatically adjusted to greater than or equal to 7.0,to enhance iodine retention in the containment water (Ref. 1). Chemical addition is necessary to counter the affects of the boric acid contained in the safety injection supplies and acids produced in the post-Loss of Coolant Accident (LOCA) environment (nitric acid from the irradiation of water and air and hydrochloric acid from irradiation and pyrolysis of electric cable insulation). The desired pH values significantly reduce formation of elemental iodine in the containment water, which reduces the production of organic iodine and the total airborne iodine in the containment. This pH adjustment is also provided to prevent stress corrosion cracking of safety related containment components during long-term cooling. Dodecahydrate trisodium phosphate (TSP) contained in four baskets provides a passive means of pH control for such accidents. The baskets are made of stainless steel with a mesh front that readily permits contact with water. These baskets are located inside containment in a trough in the heavy floor adjacent to the four IRWST strainer openings. Recirculation of water inside the containment, following a LOCA, is driven by the core decay heat and provides mixing to achieve a uniform solution pH. The dodecahydrate form of TSP (Na_3PO_4-$12H_2O$) is initially loaded into the baskets because it is hydrated and will undergo less physical and chemical change than would anhydrous TSP as a result of the humidity inside containment (Reference 1).
APPLICABLE SAFETY ANALYSIS	The LOCA radiological consequences analyses takes credit for iodine retention in the IRWST. In the event of a Design Basis Accident (DBA), iodine may be released from the fuel into the containment. To limit this iodine release from containment, the pH of the water in the IRWST is adjusted by the addition of TSP. Adjusting the IRWST to neutral or alkaline pH (pH ≥ 7.0) will augment the retention of the iodine, and thus reduce the iodine available to leak to the environment. The pH adjustment satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO	The TSP is required to adjust the pH of the recirculation water to > 7.0 after a LOCA. A pH > 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH > 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components. A required volume is specified instead of mass because it is not feasible to weigh the TSP in the containment. The minimum required volume is based on the manufactured density of TSP (58 lb/ft^3). This is conservative because the density of TSP may increase after installation due to compaction.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause release of radioactive iodine to containment requiring pH adjustment. The pH adjustment baskets assist in reducing the airborne iodine fission product inventory available for release to the environment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, pH adjustment is not required to be OPERABLE in MODES 5 and 6.
ACTIONS	<u>A.1</u> If the TSP volume in the baskets is not within limits, the iodine retention may be less than that assumed in the accident analysis for the limiting DBA. Due to the very low probability that the volume of TSP may change, the variations are expected to be minor such that the required capability is substantially available. The 72 hour Completion Time for restoration to within limits is consistent with times applied to minor degradations of ECCS parameters. <u>B.1 and B.2</u> If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

The minimum amount of TSP is 211 ft^3. This volume is based on providing sufficient TSP to buffer the post-accident containment water to a minimum pH of 7.0. Additionally, the TSP volume is based on treating the maximum volume of post-accident water (628,320 gallons) containing the maximum amount of boron (1800 ppm) as well as other sources of acid. The minimum required mass of TSP is 12,200 pounds.

The minimum required volume of TSP is based on this minimum required mass of TSP, the minimum density of TSP plus margin to account for degradation of TSP during plant operation. The minimum TSP density is based on the manufactured density, since the density may increase and the volume decrease, during plant operation, due to agglomeration from humidity inside the containment. The minimum required TSP volume also has approximately 10% margin to account for degradation of TSP during plant operation.

The periodic verification is required every 24 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 24 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the Containment Building.

REFERENCES

1. FSAR Section 6.3.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The MSSVs provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurization of the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the condenser, is not available. This is done in conjunction with the Emergency Feedwater System (EFW) providing cooling water from the EFW Storage Pools.

The MSSVs are spring-loaded safety valves. Two MSSVs are located on each Main Steam Line, outside containment, upstream of the main steam isolation valves and downstream of the Main Steam Relief Train (MSRT), as described in FSAR Section 10.3 (Ref. 1).

The MSSVs along with the MSRTs provide overpressure protection of the main steam piping and steam generators. Together, the MSSVs and MSRTs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV setpoints and capacities are such that with consideration of reactor trip, the MSSVs alone will prevent main steam pressure from rising above 110% of the steam generator design pressure upon full loss of load.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure during an anticipated operational occurrence (AOO) or postulated accident.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in FSAR Section 15.2 (Ref. 3). Of these, the closure of a single main steam isolation valve without main steam bypass or partial trip function is the limiting AOO. Closure of a single MSIV results in a smaller isolated volume on the secondary side, therefore this event is more limiting than a turbine trip event for secondary system over pressure.

The safety analysis demonstrates that the transient response for MSSV closure occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSRTs and MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. These events are bounded by the MSSV closure event.

The safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The accident analysis requires that the two MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that the two MSSVs per steam generator be OPERABLE in compliance with Reference 2 and the postulated accident analysis.

The OPERABILITY of the MSSVs are defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and be closed or reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their design safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, two MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSRTs or MSSVs to be OPERABLE in these MODES.

BASES

ACTIONS

A.1 and A.2

With one required MSSV inoperable, the associated MSRT is verified OPERABLE and action must be taken to restore the valve to OPERABLE status within 30 days. Verification of MSRT OPERABILITY is performed as an administrative check by examining logs or other information to determine if an MSRT is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the MSRT. If the OPERABILITY of the associated MSRT cannot be verified, however, Condition C must be immediately entered.

The 30 day Completion Time considers the following:

a. With one MSSV inoperable, the resulting relief capacity of the affected SG is 75% (taking into account the MSRT) of the full load steam generation of the assigned steam generator, which is greater than the 50% relief capacity considered in the safety analysis.

b. The remaining OPERABLE overpressure protection devices are sufficient for heat removal for the long term phase.

The 30 day Completion Time is considered reasonable for restoring the inoperable components to OPERABLE status.

B.1

With two MSSVs inoperable, actions must be taken to restore the inoperable MSSVs to OPERABLE status within 7 days.

This Completion Time is applicable because:

a. With two MSSVs inoperable on the same SG, the resulting relief capacity of the affected SG is 50% (taking into account the MSRT) of the full load steam generation per SG, which is equal to 50% relief capacity considered in the safety analysis.

b. With one MSSV inoperable on one SG and one MSSV inoperable on another SG, the resulting relief capacity for each SG is 75% of the full load steam generation per SG. This combination of inoperabilities is different from the one in the safety analysis. However, the total relief capacity of the four SGs is 350% of the full load steam generation per SG, which is the exact relief capacity considered in the safety analysis.

c. The remaining OPERABLE overpressure protection devices (MSRTs) are sufficient for heat removal in the long term phase.

BASES

ACTIONS (continued)

The 7 day Completion Time is considered reasonable based on operating experience to accomplish the Required Action in an orderly manner without challenging unit systems.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met or if three or more MSSVs are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code (Ref. 4) requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5).

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. The SR allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

BASES

REFERENCES

1. FSAR Section 10.3.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. FSAR Section 15.2.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. ANSI/ASME OM-1-1987.
6. FSAR Section 15.4.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located on each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and main steam relief train (MSRT), to prevent MSSV and MSRT isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, turbine bypass, and other auxiliary steam supplies from the steam generators.

The MSIVs are controlled by two redundant and parallel control lines. Each control line is composed of:

a. Two fast closure pilot valves in series actuating a common fast closure distributor; and

b. An exercise pilot valve actuating an exercise distributor.

The arrangement of pilot valves prevents a failure in any pilot valve to cause either a spurious closing (two pilot valves in series) or a failure to close (two manifolds in parallel). The MSIVs fail safe position is closed on loss of control or power supply. The pilot valves are de-energized to close the MSIVs.

The MSIVs are closed under faulted conditions by the Protection System. The MSIVs can also be closed manually. The MSIVs fail closed on loss of control or actuation power.

A description of the MSIVs is found in FSAR Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the main steam line break (MSLB) inside containment, discussed in FSAR Section 6.2 (Ref. 2). It is also affected by the accident analysis of the MSLB and feedwater line break events presented in FSAR Chapter 15 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting case for the containment analysis is the MSLB inside containment, with offsite power available, and failure of the MSIV on the affected steam generator to close. At lower power levels, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIVs contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power.

The accident analysis compares several different MSLB events against different acceptance criteria. The double-ended guillotine break of a main steam line in the valve compartment in the Safeguards Building upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The MSLB outside containment, upstream of an MSIV at hot zero power is the limiting case for a post trip return to power. The analysis includes a spectrum of break sizes, scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. The worse case single failure is a main steam relief control valve associated with one of the unaffected steam generators failed in the fully open position (Ref. 3).

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

c. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

d. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, two control lines per MSIV are OPERABLE, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 50.34 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated, when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, normally most of the MSIVs are closed and the steam generator energy is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

BASES

ACTIONS

A.1

With only one control line of one or more MSIVs inoperable in MODE 1, the affected MSIV (s) can still be closed by the other control line, however actions must be taken to restore the inoperable control line(s) to OPERABLE status in 72 hours. The 72 hour Completion Time is reasonable considering the MSIV would be closed by the OPERABLE control line in the event of an accident.

B.1

With one MSIV inoperable due to the inoperability of both control lines or reasons other than Condition A, the MSIV must be restored to OPERABLE status within 8 hours. Otherwise the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Times are reasonable based on operating experience to reach MODE 2 and to close the MSIV(s) in an orderly manner without challenging unit systems.

C.1

If Required Action A.1 or B.1 cannot be met within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition D would be entered. The completion times are reasonable based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems.

D.1 and D.2

Condition D is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is reasonable based on operating experience.

BASES

ACTIONS (continued)

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

E.1 and E.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that each MSIV and its pilot valves are OPERABLE, i.e. that it can be closed on demand. The test is performed one valve at a time using one control line only, in MODES 1 and 2, under stable plant conditions. The Surveillance Frequency of 31 days is consistent with operating experience of similar MSIVs on existing plants.

SR 3.7.2.2

This SR verifies freedom of movement of the valve stem and disk by partial valve closure and re-opening. The MSIV design allows for this test during power operation without impairing power generation and without risk of full valve closure. The Surveillance Frequency of 92 days is consistent with operating experience of similar MSIVs on existing plants.

The Frequency is in accordance with the Inservice Testing Program and is in accordance with the ASME Code (Ref. 5). This SR is modified by a Note that limits this surveillance to MODES 1 and 2.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.3

This SR verifies that MSIV closure time is within the limit assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with an owner controlled document controlled under 10 CFR 50.59 and the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs can not be full stroke tested when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program. This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.4

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MSIV testing is every 24 months on a STAGGERED TEST BASIS for each control line. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint. This SR is modified by a NOTE that requires the performance of this surveillance prior to entry into MODE 2.

REFERENCES

1. FSAR Section 10.3.
2. FSAR Section 6.2.
3. FSAR Chapter 15.
4. 10 CFR 50.34.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater (MFW) Valves

BASES

BACKGROUND

On each of the four steam generators (SGs), the Main Feedwater valves (MFW Full Load Isolation Valves (MFWFLIVs), MFW Full Load Control Valves (MFWFLCVs), MFW Low Load Isolation Valves (MFWLLIVs), FW Low Load Control Valves (MFWLLCVs), MFW Very Low Load Control Valves (MFWVLLCVs), and MFW Main Isolation Valves (MFWMIVs)) are located in valve stations, physically separated from each other and from other systems. Within these valve compartments, the MFW lines are arranged in three trains, one Very Low Load, one Low Load, and one Full Load train. The full load flow path for each steam generator includes one MFWFLCV, one MFWFLIV, and the MFWMIV. The low load flow path for each steam generator includes one MWFLLCV, one MFWLLIV, and the MFWMIV. The very low load flow path for each steam generator includes one MFWVLLCV, one MFWLLIV, and the MFWMIV. Each of these trains can be isolated redundantly by one isolation valve, one control valve, or the MFWMIV. The Low Load isolation valve allows isolation of the Low Load and the Very Low Load train at the same time.

The closure of these valves allows limiting the filling of the steam generators in case of a too high feedwater flowrate which could impair the functioning of the safety valves of the Main Steam System.

In the event of a secondary side pipe rupture inside containment, the valves also limit the quantity of high energy fluid that enters containment through the break and provide a pressure boundary for the controlled addition of Emergency Feedwater (EFW) to the intact loops. They also reduce the cooldown effects in case of Main Steam Line Breaks (MSLBs) or in case of excessive increase in feedwater flowrate caused by a feedwater system malfunction.

A MFW Isolation valve outside containment and a MFW check valve inside containment provide the containment isolation function.

The MFWFLIVs and MFWFLCVs close on a reactor trip. The low and low-low range control and isolation valves close in response to steam generator level as described in Reference 1. The MFWMIV closes on a containment isolation signal. The MFW valves may also be actuated manually.

A description of the MFW valves is found in FSAR Section 10.4.7 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES	The Full Load Line must be isolated on each of the four SGs by redundant means in case of reactor trip or a high SG level signal. The Low and Very Low Load line of the affected SG must be isolated in case of high level, low pressure or a high pressure drop signal coming from the SG. These actions are needed to mitigate the following accidents: MSLB; Feedwater Line Break (FWLB); Steam Generator Tube Rupture (SGTR); or Feedwater Malfunction. The failure of these respective valves to close could lead to an overcooling event causing re-criticality (in case of MSLB or feedwater malfunction), to increase the mass and energy releases inside containment (in case of MSLB or FWLB) or to fill the steam lines with feedwater (in case of SGTR or feedwater malfunction). The MFW valves close on reactor trip and feedwater isolation signals as described in detail in Ref. 1. Each flow path has three isolation or control valves in series in addition to a check valve located inside Containment. The MFW valves satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).
LCO	This LCO ensures that the MFW isolation and control valves will reduce or isolate MFW flow to the steam generators, as required, following an excessive feedwater flow accident, a FWLB, an SLB, or an SGTR. It also ensures that the MFWMIV provides isolation for events requiring containment isolation. This LCO requires that four MFWFLIVs, four MFWFLCVs, four MFWLLIVs, four MFWLLCVs, four MFWVLLCVs, and four MWFMIVs be OPERABLE. The MFWFLIVs, MFWFLCVs, MFWLLIVs, MFWLLCVs, MFWVLLCVs, and MFWMIVs are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal. Failure to meet the LCO requirements can result in additional mass and energy being released to containment, the introduction of water into the main steam lines, or an overcooling of the primary circuit depending on the accident considered.
APPLICABILITY	The feedwater isolation and control valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and in the steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODE 1 and in MODES 2 and 3 except when closed and de-activated, the full-load, low load, and very low load isolation and control valves are required to be OPERABLE to limit the amount of water in the steam generator, to limit the overcooling of the primary circuit, or to limit the amount of water that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated, they are already performing their safety function.

BASES

APPLICABILITY (continued)

In MODES 4, 5, and 6, steam generator energy is low and all MFW valves are normally closed since MFW is not required.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each flow path.

A.1

With one valve in the full load flow path inoperable, action must be taken to restore the affected valve to OPERABLE status within 7 days. The 7 day Completion Time takes into account the redundancy afforded by the associated MFWFLCV, MFWFLIV, and MFWMIV; and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 7 day Completion Time is reasonable, based on operating experience.

B.1

With two valves in the full load flow path inoperable, action must be taken to restore the affected valves to OPERABLE status within 72 hours. The 72 hour Completion Time takes into account the redundancy afforded by the associated MFWFLCV, MFWFLIV, and MFWMIV; and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 72 hour Completion Time is reasonable, based on operating experience.

C.1

With three valves in the full load flow path inoperable, action must be taken to restore the affected valves to OPERABLE status within 8 hours. The 8 hour Completion Time takes into account the redundancy afforded by the redundant actuation trains on MFW full load flow path valves and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 8 hour Completion Time is reasonable based on operating experience.

BASES

ACTIONS (continued)

D.1 and D.2

With one or more MFWLLIVs, MFWLLCVs, or MFWVLLCVs in the low load or very low load flow path inoperable, action must be taken to restore the affected valve to OPERABLE status within 8 hours or isolate the flow path. When the valves are closed, they are performing their required safety function

Inoperable MFW low load and very low load flow path valves that are closed as a result of this Required Action, must be verified on a periodic basis that they are closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

E.1 and E.2

If the MFWs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFWFLIV, MFWFLCV, MFWLLIV, MFWLLCV, MFWVLLCV, and MFWMIV is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time in accordance with an owner controlled document controlled under 10 CFR 50.59 and the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.2

This SR verifies that each valve can close on an actual or simulated actuation signal. This Surveillance is normally performed during shutdown or upon returning the plant to operation following a refueling outage.

The Frequency for this SR is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 10.4.7.
2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.4 Main Steam Relief Trains (MSRTs)

BASES

BACKGROUND

The MSRTs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the condenser not be available. This is done in conjunction with the Feedwater or Emergency Feedwater System. The MSRT valves also provide secondary overpressure protection.

One MSRT is provided for each steam generator, outside containment, upstream of the Main Steam Safety Valves and the main steam isolation valves. Each MSRT consists of one main steam relief control valve (MSRCV) located downstream of one main steam relief isolation valve (MSRIV).

The main steam relief control valves are motorized control valves, normally open, which allow control of the steam generator steam pressure, and consequently control of the cooldown rate. The MSRCVs provide a means of controlling MSRT steam flow to prevent overcooling the RCS. The MSRCVs allow mitigation of the effects of a stuck open MSRIV. The MSRCVs are automatically positioned based on thermal power.

The main steam relief isolation valves are angle globe valves with a motive steam-operated piston actuator, operated by two parallel sets of two pilot valves in series. The arrangement of pilot valves prevents a failure in any pilot valve from causing either a spurious opening (two pilot valves in series) or a failure to open (two sets of pilot valves in parallel). The MSRIVs close (fail safe position) on loss of power supply or on loss of Instrumentation and Control. The pilot valves must be energized to open the associated MSRIV.

The MSRIVs are normally closed, with the pilot valves kept closed (de-energized). The valves open automatically and quickly on demand from the Protection System.

A description of the main steam relief control valves and of the main steam relief isolation valves is found in FSAR Section 10.3 (Ref. 1)

Each MSRT minimum required capacity is 50% of the full steam generation of the assigned steam generator (for a design core power level of 4590 MWth), at a design pressure of 1,435 psig, thus limiting the system pressure to ≤ 110% of the steam generator design pressure, in order to meet the requirements of the ASME Code, Section III (Ref. 2). The minimum required capacity, combined with the MSSV capacity, provides 100% flow relief at steam generator design pressure per SG.

BASES

BACKGROUND (continued)

Each MSRT maximum capacity is limited to 50% of the full load steam generation of its assigned steam generator (at a design core power of 4590 MWth), at design pressure of 1,435 psig, thus limiting the consequences of MSRIV spurious opening with regards to reactor coolant system overcooling and reactivity control.

The MSRTs are actuated automatically by the Protection System, but can be controlled manually by the operator.

APPLICABLE SAFETY ANALYSES

The design basis of the main steam relief valves is established by the capability to remove residual heat in a controlled manner and to cool down the plant to RHRS entry conditions at various rates (normal cooldown at 90°F/h, partial cooldown at 180°F/h). The design rate of partial cooldown is applicable to events with two steam generators OPERABLE, each with one MSRT OPERABLE.

The MSRTs residual heat removal function in the safety analysis of Reference 3 is required:

a. To perform residual heat removal at the controlled rate following Condition II, III, and IV events with the condenser inoperable;

b. To perform cooldown to RHR entry conditions, following Condition II, III, and IV events with the condenser inoperable; and

c. To perform Partial Cooldown of the unit at a rate of 180°F/h from MODE 3 to 870 psia to allow Medium Head Safety Injection (MHSI) into the Reactor Coolant System in the event of a Loss of Coolant Accident (LOCA) or Steam Generator Tube Rupture (SGTR).

The Main Steam Relief Trains do not directly participate in the reactivity control function. Nevertheless, reactivity control is supported by isolating a spuriously open MSRT to limit RCS cooling. Excessive increase in steam flow causes overcooling of the reactor coolant and thus reactivity feedback to the core.

In case of a SGTR, the MSRT participates in the confinement of radioactive material. In the SGTR mitigation process, an increase in the MSRV setpoint of the affected SG over the MHSI delivery pressure enables termination of the leak flow. It also prevents overfilling of the affected SG. In the event that the condenser in inoperable, the MSRT challenge avoids response of the associated MSSVs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MSRT participates in the limitation of SG pressure increase in the most limiting Category II overpressure transient, inadvertent closure of one MSIV, thus limiting the SG pressure peak to less than 110% of the design pressure, without MSSV challenge. In other overpressure transients (Category III and IV), the MSRT participates in the limitation of the pressure peak in conjunction with the associated MSSV (see B 3.7.1). For the most limiting overpressure transient of Category 4 (i.e. loss of secondary side heat sink at full power without reactor trip), the inoperability of one MSRT is considered in the safety analysis and all other overpressure protection devices (other MSRTs and all MSSVs) are challenged and thus must be operable.

In all analyzed events, two steam generators and consequently both their MSRTs are required for residual heat removal and plant cooldown, considering a single failure of one MSRT and preventive maintenance performed on either electrical division or emergency diesel at the moment of the accident with assumed Loss of Offsite Power.

In events analyzed in Reference 3, the MSRT ensures residual heat removal by performing either Partial Cooldown or Fast Cooldown, either by automatic or manual action, depending on the event.

The MSRV position is automatically controlled by the Protection System as a function of power level, provided that the MSRIV is closed. The MSRV position is such that:

a. Consequences of a spurious MSRT event are limited with regards to the Reactor Coolant System; and

b. Mitigation of overpressure transients is ensured.

If the MSRIV opens, the MSRV is automatically switched into SG pressure control mode.

The MSRT valves satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Four main steam relief trains (MSRIVs and MSRCVs) are required to be OPERABLE so as to ensure residual heat removal by a minimum of two steam generators even in the case of preventative maintenance and a single failure affecting the other two steam generators and connected cooling systems (e.g., Emergency Feedwater System, MSRT).

Isolation capability is also required on the four MSRTs, since any steam generator can be affected by a spuriously opened MSRIV or by an SGTR.

BASES

LCO (continued)

Failure to meet the LCO either results in the inability to perform residual heat removal and plant cooldown following an event with an inoperable condenser, or in the inability to isolate a SG affected by an SGTR or a spurious MSRIV opening.

A main steam relief control valve is considered OPERABLE when it is capable of full opening and closing and when it is capable of providing controlled relief of the main steam flow, with support of related I&C systems.

A main steam relief isolation valve is considered OPERABLE when it is capable of opening and when it is capable of re-closure after challenge.

APPLICABILITY

In MODES 1, 2, and 3, the MSRTs are required to be OPERABLE to provide a decay heat removal path in conjunction with the Emergency Feedwater System.

In MODE 4, 5, or 6, decay heat removal is provided by the Low Head Safety Injection System.

ACTIONS

A.1

With one control line inoperable for opening in one or more MSRIVs (i.e. one pilot valve is blocked closed), the affected MSRIVs are still OPERABLE, however the control line(s) must be restored to OPERABLE status in 30 days. This completion time is based on the following:

a. Redundancy for MSRIV opening is provided by the second control line.

b. In case of an event with loss of the condenser and assuming a single failure on the second control line of one MSRIV to open, the residual heat removal can still be ensured by the other MSRTs.

c. In case of an overpressure event and assuming a single failure of the second control line of one MSRIV to open which leads to failure to open of the associated MSRIV, the redundancy provided by the two associated OPERABLE MSSVs ensure the pressure limitation in the affected SG.

BASES

ACTIONS (continued)

In case one pilot valve is blocked open in at least one control line of one or more MSRIVs, the isolation function of the MSRIV is not assured. The control line(s) must be restored to OPERABLE status in 30 days. This Completion Time is based on redundancy for MSRIV closure provided by the second pilot valve in series and by the MSRCV.

B.1 and B.2

With one or two MSRIVs inoperable for opening (e.g., due to a mechanical failure or due to two pilots in parallel blocked closed), the residual heat removal function and the overpressure protection function of the corresponding MSRT are not assured. The associated MSSV must be verified OPERABLE and the valves must be restored to OPERABLE status in 7 days. Verification of MSSV OPERABILITY is performed as an administrative check by examining logs or other information to determine if an MSSV is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the MSSV. If the OPERABILITY of the associated MSSV cannot be verified, however, Condition C must be immediately entered.

The Completion Time of 7 days is reasonable because:

a. In case of an event with loss of condenser, the two OPERABLE MSRTs are capable of performing the residual heat removal function. However the single failure criterion may not be met.

b. In case of an overpressure event, the redundancy provided by the two OPERABLE MSSVs ensure the limitation of SG pressure (the necessary relief capacity being 50% of full load steam generation per SG). However the single failure criterion is not met.

With one or two MSRCVs inoperable, the residual heat removal function and the overpressure protection of the corresponding MSRT are not assured, as well as the isolation function. The single failure criterion is not fulfilled any more, so the same Completion time of 7 days applies to restore MSRCV(s) to OPERABLE status.

Finally, with one or two MSRIVs inoperable for closing (e.g., blocked open during residual heat removal with MSRTs or due to a failed test), the residual heat removal function is still ensured, but the redundancy for MSRT isolation is lost because it can only be ensured by associated MSRVs. As a result, the same Completion Time of 7 days applies to restore to OPERABLE status.

BASES

ACTIONS (continued)

C.1 and C.2

If required Action A.1 or B.1 cannot be met within the required Completion Times, the unit must be placed in a MODE in which the LCO does not apply, and in which the inoperable MSRCV or MSRIVs can be restored to OPERABLE status. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

With three or more MSRIVs inoperable for opening, or three or more MSRCVs inoperable, the residual heat removal function and the overpressure protection function of the corresponding MSRT are not assured. Only one MSRT remains OPERABLE for this function, which is less than the needed two MSRTs for residual heat removal following analyzed events.

The unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4, without reliance upon steam generators for heat removal, within 12 hours.

The MSRCVs inoperability also affects the MSRT isolation function by loss of redundancy (only MSRIVs can ensure the function).

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies each MSRIV OPERABLITY by opening the valve and then by closing the MSRIV. This SR is performed once every refueling outage on a STAGGERED TEST BASIS for each control line (i.e. twice per MSRIV) in hot shutdown conditions. The frequency is reasonable based on the fact that complete opening of an MSRIV is not possible during power operation and on the operating experience of similar MSRIVs on existing plants.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.2

This SR verifies each MSRCV OPERABILITY by stroking the valve through a full cycle. The test can be performed during power operation under stable conditions without impairing power operation because the MSRIV stays closed during the test. The test can also be performed in hot shutdown conditions before plant shutdown. The frequency of once per cycle is reasonable based on operating experience and on the fact that the MSRCV operates under the control of the Protection System during power operation, which ensures that the valve is not blocked in a specific position.

SR 3.7.4.3

This SR demonstrates that each MSRIV actuates on an actual or simulated steam pressure setpoint signal. The 24 month frequency is based on the need to perform the test during either hot or cold shutdown conditions. The frequency is reasonable based on the fact that opening a MSRIV is not possible during power operation and on operating experience of similar MSRIVs on existing plants.

SR 3.7.4.4 and 3.7.4.5

This SR demonstrates that each MSRCV is automatically positioned based on thermal power and is switched into SG pressure control mode on an actual or simulated MSRIV opening. The test can be performed in hot shutdown conditions before plant shutdown. The frequency of once per cycle is reasonable based on operating experience and on the fact that the MSRCV operates under the control of the Protection System during power operation, which ensures that the valve is not blocked in a specific position.

REFERENCES

1. FSAR Section 10.3.
2. ASME Code for Operation and Maintenance of Nuclear Power Plants.
3. Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System on low Steam Generator (SG) level or Loss of Offsite Power. The EFW pumps take suction through a common supply header from their respective EFW storage pool (SP) and normally pump to their respective steam generator secondary side via separate and independent connections. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or main steam relief valves (MSRVs) (LCO 3.7.4). If the main condenser is available, steam may be released via the turbine bypass valves.

The EFW System consists of four motor driven EFW pumps and four EFW SPs configured into four separate trains. The inventory of the four EFW SPs is available to all EFW pumps through the common supply header.

The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each EFW pump is powered from an independent Class 1E power source.

The non-safety Startup and Shutdown System (SSS) is used for supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The EFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the EFW System supplies sufficient water to cool the unit to Low Head Safety Injection (LHSI) entry conditions, with steam released through the main steam relief valves.

The EFW System actuates automatically on low steam generator water level signal generated by the Protection System (LCO 3.3.1). The system also actuates on loss of offsite power signal.

The EFW System is discussed in FSAR Section 10.4.9 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The EFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the EFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3%.

In addition, the EFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

There are four EFW trains. Each EFW train has a separate SP. All four EFW SPs, the common supply and discharge headers, and the four injection paths are required to be OPERABLE. One EFW pump train is assumed to be unavailable due to maintenance and a second EFW pump train or its normal injection pathway is assumed to be lost to a single failure. Note, an EFW Pump Train includes the pump, discharge check valve, flow control valve, and piping to the manual isolation valves on the suction and discharge of the pump.

The two remaining EFW trains provide sufficient flow for decay heat removal as required by the accident analysis. For certain sized feedwater line breaks, one of the remaining EFW pumps feeds a faulted steam generator. This pump is re-aligned from the MCR at 30 minutes to feed through the injection pathway associated with the train whose pump is unavailable due to maintenance.

The limiting accident for the EFW System is a Main Feedwater Line Break (MFWLB) with a natural circulation cooldown.

In addition, the minimum available EFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident (LOCA).

The Protection System automatically actuates the EFW pumps and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power.

The EFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(d)(2)(ii) for operation in MODES 1, 2, and 3 and Criterion 4 of 10 CFR 50.36(d)(2)(ii) for operation in MODE 4.

BASES

LCO	This LCO provides assurance that the EFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Four EFW pumps, the common supply and discharge headers, and the four injection paths are required to be OPERABLE to ensure decay heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering each of the pumps from independent emergency buses. The EFW System is configured into four trains, which share common supply and discharge headers. The EFW System is considered OPERABLE when the components and common flow paths required to provide redundant EFW flow to the steam generators are OPERABLE. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE. In MODE 4 with only one EFW pump OPERABLE, operation is allowed to continue because only one EFW pump is required in accordance with the Note that modifies the LCO. Because of the reduced heat removal requirements and the short period of time in MODE 4, one EFW pump is sufficient to remove decay heat. Although not required, the unit may continue to cool down to LHSI entry conditions.
APPLICABILITY	In MODES 1, 2, and 3, the EFW System is required to be OPERABLE in the event that it is called upon to function when MFW and offsite power are lost. In addition, the EFW System is required to supply enough makeup water to replace the secondary inventory, lost as the unit cools to MODE 4 conditions. In MODE 4 and 5, the EFW System may be used for heat removal via the steam generators. In MODE 6, the steam generators are not normally used for heat removal, and the EFW System is not required
ACTIONS	A Note prohibits the application of LCO 3.0.4.b for two or more EFW trains inoperable when entering MODE 1. There is an increased risk associated with entering MODE 1 with two or more EFW trains inoperable and the provisions of LCO 3.0.4.b, which allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

BASES

ACTIONS (continued)

A.1

With one EFW train inoperable in MODE 1, 2, or 3, action must be taken to restore OPERABLE status within 120 days. The 120 day Completion Time is reasonable, based on the FSAR Chapter 15 analysis assumption that one EFW train is not available due to maintenance, and the low probability of a postulated accident occurring during this time period.

B.1

With two EFW trains inoperable in MODES 1, 2, or 3, action must be taken to restore at least one inoperable EFW train to OPERABLE status in 72 hours. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of a postulated accident occurring during this time period.

C.1 and C.2

When any Required Action and associated Completion Time cannot be met; or if three EFW trains are inoperable in MODE 1, 2, or 3; or the common injection header; the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

With four EFW trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown and only non-safety means for conducting a cooldown with the SSS. In such a condition, the unit should not be perturbed by any action, including a power change that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one EFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one EFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

BASES

ACTIONS (continued)

E.1

In MODE 4, either the reactor coolant pumps or the LHSI loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the one required EFW pump inoperable, action must be taken to immediately restore an inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW System flow paths provides assurance that the proper flow paths will exist for EFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. For the EFW System, this SR includes the steam generator blowdown isolation valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Each EFW pump suction and supply header isolation valve is required to be locked open at 31 day intervals. This surveillance is designed to ensure that all EFW pumps can the inventory of all EFW pools.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.3

Each EFW discharge header cross-connect valve is required to be cycled in order to assure the capability for any EFW pump to feed any steam generator as assumed in the main feedwater line break (Ref. 3) The Frequency of this SR is in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 2). Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

SR 3.7.5.5

This SR verifies that EFW can be delivered to the appropriate steam generators in the event of any accident or transient that generates a Protection System actuation, by demonstrating that each automatic valve in the flow path actuates to its correct position, each EFW pump starts automatically, and flow rate is controlled within required limits and steam generator level is controlled within limits, on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

SR 3.7.5.6

This SR verifies that the EFW is properly aligned by verifying the flow paths from the supply header to its respective steam generator prior to entering MODE 2 after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of EFW flow paths must be verified before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the SP to the steam generators is properly aligned.

BASES

REFERENCES

1. FSAR Section 10.4.9.
2. ASME Code for Operation and Maintenance of Nuclear Power Plants.
3. FSAR Section 15.2

B 3.7 PLANT SYSTEMS

B 3.7.6 Emergency Feedwater (EFW) Storage Pools

BASES

BACKGROUND

The EFW pumps take suction through separate suction lines from their respective EFW storage pool (SP) and normally pump to their respective steam generator secondary side via separate and independent connections. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or main steam relief trains (MSRTs) (LCO 3.7.4). If the main condenser is available, steam may be released via the steam bypass valves.

The EFW System consists of four motor driven EFW pumps and four EFW SPs configured into four separate trains. The inventory of the four EFW SPs is available to all EFW pumps through the common supply header.

Because the SPs are principal components in removing residual heat from the Reactor Coolant System (RCS), they are designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The SPs are designed to Seismic Category I to ensure availability of the feedwater supply. A description of the SPs is found in FSAR Section 10.4.9 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The EFW SPs provide cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in Chapters 6 and 15 (Ref. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally four hours at MODE 3, steaming through the MSSVs and MSRVs followed by a cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate or a lower cooldown rate if offsite power is not available.

The limiting accident for the EFW SPs is a Main Feedwater Line Break (MFWLB) with a natural circulation cooldown.

The EFW SPs satisfy the requirements of Criterion 2 and 3 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO	To satisfy accident analysis assumptions, the EFW SPs must contain sufficient water to remove decay heat for four hours following a reactor trip from 102% RTP and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the EFW pumps during cooldown or before isolating EFW to a faulted steam generator. The EFW SP required usable volume of 300,000 gallons is based on a cooldown to RHR entry conditions at 50°F/hour, with all four reactor coolant pumps in service. This basis is established in Reference 1 and exceeds the volume required by the accident analysis. The OPERABILITY of the EFW SPs is determined by summing the available tank volumes. The volume in an SP is considered usable when it is aligned to the common supply header.
APPLICABILITY	In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the EFW SPs are required to be OPERABLE to support EFW System operability. In MODE 5 or 6, the EFW SPs are not required because the EFW System is not required.
ACTIONS	<u>A.1 and A.2</u> With one EFW SPs inoperable in MODE 1, 2, or 3, or MODE 4, when a steam generator is being relied upon for heat removal, action must be taken to verify the usable volume in the remaining SPs is ≥ 300,000 gal. and to declare the associated EFW train inoperable. <u>B.1 and B.2</u> With two or more EFW SPs inoperable or the usable volume of the available SPs is < 300,000 gal., the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4, without reliance on a steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner, and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.6.1 This SR verifies that the EFW Storage Pools contain the required volume of cooling water. The 24 hour Frequency is based on operating experience and are not used by other systems and that the SPs have no other function that to supply water to the EFW trains. Also, the 24 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the SP levels. SR 3.7.6.2 This SR verifies every 31 days that the EFW supply cross connect valves are locked open. This verification ensures that the usable volume in the SPs are available to all EFW trains through the supply cross connect header and ensures timely discovery if a valve should be not locked open. If an EFW supply cross connect valve is not open, the usable volume of the SP is not available to each of the four EFW trains as assumed in the safety analysis. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned EFW supply cross connect valve is unlikely.
REFERENCES	1. FSAR Section 10.4.9. 2. FSAR Chapter 6. 3. FSAR Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND	The CCW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Service Water (ESW) System and thus to the environment. The CCW System consists of four separate safety classified trains (1, 2, 3 and 4) corresponding to the four layout divisions (1, 2, 3 and 4) and two separate common headers. One of the common headers (common 1 header) is connected normally either to train 1 or to train 2. The other common header (common 2 header) is connected either to train 3 or to train 4. A set of isolation valves per train can separate each train from the common header and either common header is capable of providing safety related cooling of the reactor coolant pump (RCP) thermal barrier cooling common loop. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal from the Protection System, and all nonessential components are isolated. Additional information on the design and operation of the system, along with a list of the components served, is presented in FSAR Section 9.2.2 (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the safety related systems and operational cooling loads to the heat sink via the ESW System. This may be during a normal or post accident cooldown and shutdown.
APPLICABLE SAFETY ANALYSES	The design basis of the CCW System is for two CCW trains to remove the post loss of coolant accident (LOCA) heat load from the In-containment Water Storage Tank (IRWST) by cooling the Low Head Safety Injection System heat exchanger at a maximum CCW temperature of 113°F (Ref. 2). The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA each model the minimum performance of the CCW System, respectively. During a unit cooldown to MODE 5 (T_{cold} < 200°F), a maximum temperature of 113°F is assumed. This maintains the IRWST fluid within acceptable limits following a LOCA as it is supplied to the Reactor Coolant System by the ECCS pumps.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The CCW System also functions to cool the unit from residual heat removal (RHR) entry conditions (T_{cold} < 350°F), to MODE 5 (T_{cold} < 200°F), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR loops operating. Two CCW trains are sufficient to remove decay heat during subsequent operations with T_{cold} < 200°F. This assumes a maximum service water temperature of 95°F occurring simultaneously with the maximum heat loads on the system.

To meet single failure criteria for the RCP thermal barrier cooling function, the load is required to be cooled by a common header which is capable of being connected two OPERABLE CCW trains. A single failure of a train initiates an automatic system response to transfer the common header to the remaining train.

The CCW System satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The CCW System consists of four trains. Four CCW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

A CCW train is considered OPERABLE when:

a. The pump and associated surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

With the exception of the RCP thermal barrier cooling common loop, the isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the Low Head Safety Injection heat exchanger.

BASES

APPLICABILITY (continued)

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A Note has been added to indicated that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable CCW train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

A.1

Required Action A.1 is modified by a Note indicating that the Required Action of A.1 is not applicable if CCW trains are inoperable in both common headers. In this condition, the RCP thermal barrier cooling common loop cannot be aligned to common header capable of being connected to two OPERABLE trains.

If one CCW train is inoperable, action must be taken to align the RCP thermal barrier cooling common loop to a common header capable of being supplied by two OPERABLE CCW trains within 72 hrs. In this condition, the CCW System can perform the RCP thermal barrier cooling function given a single failure. The 72 hour Completion Time is reasonable, based on the low probability of a postulated accident occurring during this period.

A.2

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 120 days. In this condition, the remaining OPERABLE CCW trains are adequate to perform the heat removal function.

B.1

If two CCW trains are inoperable, action must be taken to restore one train to OPERABLE status within 72 hours. In this condition, the remaining OPERABLE CCW trains are adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE trains, and the low probability of a postulated accident occurring during this period.

BASES

ACTIONS (continued)

C.1 and C.2

If a CCW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed

Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components, other than the RCP thermal barrier cooling common loop, may render those components inoperable but does not affect the OPERABILITY of the CCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to

BASES

SURVEILLANCE REQUIREMENTS (continued)

perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 9.2.2.
2. FSAR Section 6.2.

B 3.7 PLANT SYSTEMS

B 3.7.8 Essential Service Water (ESW) System

BASES

BACKGROUND

The ESW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, and a normal shutdown, the ESW System also provides this function for the associated safety related and nonsafety related systems. The safety related function is covered by this LCO.

The ESW System consists of four separate safety related, cooling water trains. Each train consists of one mechanical draft cooling tower, associated basin, pump, piping, valving, instrumentation, and mechanical filtration. Each safety related 2-cell seismic Category I mechanical draft cooling tower rejects energy from the ESW fluid to the ambient and returns the cooled fluid to the ESW cooling tower basin, from which the ESW pumps take suction. Each ESW cooling tower basin is sized for 3 days of post loss of coolant accident (LOCA) operation and ensures adequate volume for the required net positive suction head (NPSH) for the associated ESW pump. Post LOCA evaporative losses are replenished by a safety related seismic Category I source of makeup water. The train associated safety related make-up source delivers water to each basin at ≥ 300 gpm to maintain the NPSH for the ESW pump for up to 30 days following a LOCA. The system pumps and valves are remote and manually aligned, except in the unlikely event of a LOCA or loss of offsite power. The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions.

The mechanical draft cooling towers and basins are safety related, seismic Category I structures sized to provide heat dissipation for safe shutdown following an accident. The cooling tower is protected from tornado missiles.

The seismic Category 1 emergency makeup water supply, to the ESW cooling tower basins, necessary to support 30 days of post accident mitigation is provided by the safety-related Ultimate Heat Sink (UHS) Makeup Water System that draws water from Lake Ontario. Lake Ontario water enters the Intake Structure forebay through two intake tunnels. The UHS Makeup Water System portion of the Intake Structure houses four independent UHS Makeup Water System trains, one for each ESW division. Each train has one pump, a discharge check valve, and a pump

BASES

BACKGROUND (continued)

discharge isolation motor operated valve, all housed in the UHS Makeup Water Intake Structure, plus the buried piping running up to and into the ESW pumphouse at the ESW cooling tower basin. Each UHS Makeup Water System pump is rated at 750 gpm.

Additional information about the design and operation of the ESW System along with a list of the components served, is presented in FSAR Section 9.2.1 (Ref. 1). The principal safety related functions of the ESW System is the removal of decay heat from the reactor and reactor coolant pump thermal barrier cooling via the Component Cooling Water (CCW) System and removal of operational heat from the emergency diesel generator (EDG).

APPLICABLE SAFETY ANALYSES

The design basis of the ESW System is for two ESW trains, in conjunction with the CCW System, to remove core decay heat and support containment cooling following a design basis LOCA as discussed in FSAR Section 6.2 (Ref. 2). This maintains the In-containment Water Storage Tank fluid within acceptable limits following a LOCA as it is supplied to the Reactor Coolant System by the Emergency Core Cooling System pumps. The ESW System also provides cooling to the train EDG during an anticipated operational occurrence (AOO) or postulated accident.

The ESW System, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR), as discussed in FSAR Section 5.4.7 (Ref. 3), entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CCW and RHR loops that are operating. Two ESW trains are sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ESW System temperature of 95°F occurring simultaneously with maximum heat loads on the system.

Each ESW basin is sized for 3 days of post LOCA operation without requiring makeup. ESW basin makeup is required to maintain NPSH for the ESW pumps beyond 3 days. This volume of water is assumed to be at ≤ 90°F during normal plant operation to prevent exceeding the maximum ESW temperature during a LOCA.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure. The ESW cooling tower and basin is designed in accordance with Regulatory Guide 1.27 (Ref. 4), which requires a 30 day supply of cooling water in the ESW basin, or equivalent make-up.

The ESW System satisfies Criterion 2 and 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The ESW System consists of four trains. Four ESW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

An ESW train is considered OPERABLE when two cooling tower fans, pump, associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE and the ESW basin contains ≥ 27.2 feet of water at ≤ 90°F with capability from makeup from OPERABLE source. An OPERABLE emergency makeup water source consists of one OPERABLE train of the UHS Makeup Water System capable of providing makeup water to its associated ESW cooling tower basin. Each UHS Makeup Water System train includes a pump, valves, piping, instruments and controls to ensure the transfer of the required supply of water from Lake Ontario to its associated ESW cooling tower.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ESW System and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ESW System are determined by the systems it supports.

BASES

ACTIONS

The actions have two Notes added. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable ESW train results in an inoperable EDG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ESW train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

A.1

If one ESW train is inoperable, action must be taken to restore OPERABLE status within 120 days. In this condition, the remaining OPERABLE ESW trains are adequate to perform the heat removal function.

The 120 day Completion Time to restore an ESW train to OPERABLE is reasonable since its operation is not assumed in the safety analysis to mitigate the consequences of postulated accidents or AOOs, it provides a reasonable time for repairs, and the low probability of a postulated accident or AOO occurring during this period.

B.1

If two ESW trains are inoperable, action must be taken to restore one to OPERABLE status within 72 hours. In this condition, the two remaining OPERABLE ESW train are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in one of the OPERABLE ESW trains could result in loss of ESW System function.

The 72 hour Completion Time is based on the redundant capabilities afforded by the two OPERABLE trains, and the low probability of a postulated accident occurring during this time period.

C.1 and C.2

If an ESW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that adequate short term (3 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the ESW pumps during the first 3 days post LOCA. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the ESW basin water level is ≥ 27.2 feet from the bottom of the basin.

SR 3.7.8.2

This SR verifies that the ESW System is available to cool the CCW System and EDG to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a postulated accident. With water temperature of the ESW basin ≤ 90°F, the design basis assumption associated with initial ESW temperature are bounded. With the water temperature of the ESW basin > 90°F, long term cooling capability of the Emergency Core Cooling System (ECCS) loads and Diesel Generators (DGs) may be affected. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.8.3

This SR is modified by a Note indicating that the isolation of the ESW components or systems may render those components inoperable, but does not affect the OPERABILITY of the ESW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the ESW flow path provides assurance that the proper flow paths exist for ESW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.4

Operating each cooling tower fan for ≥ 15 minutes in all speed settings verifies that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the ESW cooling tower fans occurring between surveillances.

SR 3.7.8.5

This SR verifies proper automatic operation of the ESW valves on an actual or simulated actuation signal. The ESW System is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.6

This SR verifies proper automatic operation of the ESW pumps and cooling tower fans on an actual or simulated actuation signal. The ESW System is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.7

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified makeup flowrate ensures that sufficient NPSH can be maintained to operate the ESW pumps following the first 3 days post LOCA. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. This SR verifies that the ESW makeup flowrate is ≥ 300 gpm.

REFERENCES

1. FSAR Section 9.2.1.
2. FSAR Section 6.2.
3. FSAR Section 5.4.7.
4. Regulatory Guide 1.27.

B 3.7 PLANT SYSTEMS

B 3.7.9 Safety Chilled Water (SCW) System

BASES

BACKGROUND

The SCW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, and a normal shutdown, the SCW System also provides this function for the associated safety related systems. The safety related function is covered by this LCO.

The SCW System consists of four independent trains. Each train consists of a chiller refrigeration unit (three 50% compressors per unit), chilled water pumps (two 100% pumps), surge tank, piping, valving, and instrumentation. Heat is rejected to the system chilled water as it passes through the cooling coils of the system users. This heat is rejected from the system as it is pumped through the train chiller refrigeration units. Trains 1 and 4 reject this energy to ambient via air cooled condensers while trains 2 and 3 have condensers cooled by the Component Cooling Water (CCW) System.

The SCW System is normally operating and cools the Control Room Air Conditioning System (CRACS), Safeguards Building Ventilation System Electrical Division (SBVSED), and the train 1 and 4 Low Head Safety Injection (LHSI) pump motor and seal coolers. The combined HVAC function of the SBVSED and SCW systems is backed by a non-safety related, 100% capacity maintenance train which is cooled by the Operational Chilled Water System.

Following a loss of offsite power, previously running SCW trains return to operation once the emergency diesel generator is started and the associated AC electrical power division is re-energized.

The SCW System operation is discussed in FSAR Section 9.2.8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the SCW System is to provide chilled water as a heat sink for the CRAC and SBVSED safety-related HVAC Systems in addition to the LHSI pump motor and seal coolers (train 1 and 4 only). This supports maintaining an acceptable environment in the main control room (MCR) and for safety-related equipment in the essential rooms housing electrical, Instrumentation and Control System, Emergency Feedwater System, and CCW System equipment in the Safeguard Buildings as well as supporting the long term operation of the cooled LHSI pumps in the event of an AOO or postulated accident. Cooling of the electrical rooms requires the availability of each train of SCW in order to ensure the ability of the plant to meet all required safety related functions during any AOO or postulated accident.

BASES

APPLICABLE SAFETY ANALYSES (continued)

A single active failure of a component of the SCW System, with a loss of offsite power, does not impair the ability of the system to perform its design function. The SCW System is designed in accordance with Seismic Category I requirements.

The SCW System satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The SCW System consists of four trains. Four SCW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

An SCW train is considered OPERABLE when one pump, surge tank, the chiller refrigeration unit with two compressors, associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE and in operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SCW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SCW System and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the SCW System are determined by the systems it supports.

ACTIONS

A.1

If one SCW train is inoperable, action must be taken to restore to OPERABLE status within 72 hours. In this condition, the three remaining OPERABLE SCW trains are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in one of the OPERABLE ESW train could result in loss of SCW System function.

The 72 hour Completion Time is based on the redundant capabilities afforded by the three OPERABLE trains, and the low probability of a postulated accident occurring during this time period.

BASES

ACTIONS (continued)

B.1 and B.2

If the SCW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR requires verification every 24 hours that each SCW train is in operation. Verification includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 24 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor SCW train performance.

SR 3.7.9.2

This SR is modified by a Note indicating that the isolation of the SCW components or systems may render those components inoperable, but does not affect the OPERABILITY of the SCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the SCW flow path provides assurance that the proper flow paths exist for SCW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.3

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the in the control room heat load calculation. This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the SCW system is slow and is not expected over this time period.

SR 3.7.9.4

This SR verifies proper automatic operation of the SCW train on an actual or simulated actuation signal. The SCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 9.2.8.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Filtration (CREF)

BASES

BACKGROUND

The CREF provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity or smoke.

The CREF consists of two 100% capacity iodine filtration trains which operate when radioactive contamination is detected at the site or inside the control room envelope (CRE) area. The iodine filtration train is a bypass path of the fresh air intake train for the Control Room Air Conditioning System (CRACS) normal air supply. The air from CRE can also be recirculated through the CREF Iodine Filtration trains. The iodine filtration trains are provided as bypass lines on two of the four normal CRACS air intake trains; other two CRACS intake trains do not have the bypass iodine filtration trains. During an emergency, the fresh outside air and recirculated air are directed through air intake motorized damper and electric heater through the CREF Iodine Filtration train. Each iodine filtration train consists of motorized damper, electric heater, prefilter, upstream HEPA filter, an activated carbon iodine filter, downstream HEPA filter, booster fan, and manual isolation damper. The filtered and clean air is then directed through one or both CRACS normal 75% capacity air conditioning train. Each air conditioning train consists of volume control manual damper, cooling coil, moisture separator, fan suction and discharge silencers, supply air fan, HEPA filter, steam humidifier, non-return damper, volume control electric damper, and fire dampers. The conditioned and clean air is then supplied to the CRE areas. Electric heaters are installed in the CRE supply air ducts to maintain individual room temperatures and relative humidity. The exhaust air from the CRE areas is directed through the recirculation air shaft and then recycled either through the iodine filtration trains or CRACS air conditioning trains. The exhaust from kitchen and sanitary areas is separated from the recycle return air and processed separately.

The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and carbon adsorbers. The HEPA filter bank downstream of the carbon iodine filter collects carbon fines and provides backup in case of failure of the upstream HEPA filter bank. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and carbon adsorbers.

BASES

BACKGROUND (continued)

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CREF train is an emergency system, which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), the outside fresh air supply to the CRE is isolated, and the outside air is directed through the CREF train. The CRE ventilation air is recycled through the air conditioning filter trains and/or CREF train.

Actuation of the CREF places the system in the emergency radiation mode of operation. Actuation of the system to the emergency radiation mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the CREF trains. The emergency radiation state also maintains control room pressurization and filtered ventilation of the air supply to the CRE.

Outside makeup air is supplied through the iodine filtration train and added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary.

The outside air entering the CRE is continuously monitored by radiation detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic gas isolation state, as required.

BASES

BACKGROUND (continued)

One CREF operating at a flow rate of < 4000 cfm will pressurize the CRE to ≥ 0.125 inches water gauge relative to all external areas adjacent to the CRE boundary. The CREF operation in maintaining the CRE habitability is discussed in FSAR Section 9.4.1 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across one of the other filter trains. Normally open isolation dampers are arranged in series so the failure of one damper to shut will not result in a breach of isolation. The CREF train components are designed in accordance with Seismic Category I requirements.

The CREF is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a postulated accident without exceeding a 5 rem whole body dose or its equivalent to any part of the body 5 rem total effective does equivalent (TEDE).

APPLICABLE SAFETY ANALYSES

The CREF components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREF provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in Chapter 15 (Ref. 2).

The CREF consists of two 100% capacity iodine filtration trains. Each iodine filtration train can be aligned with one of the two 75% capacity air conditioning trains. There are only two iodine filtration trains since only slow failure modes are assumed and filtration efficiency is checked periodically. Both CREF trains with the associated air conditioning trains are required to be OPERABLE. One CREF train is assumed to be lost to a single failure. The other train provides 100% of the ventilation to the CRE.

The CREF provides protection from smoke to the CRE occupants. Reference 3 discusses that the need for protection of CRE occupants following a hazardous chemical release is not required at NMP3NPP. Reference 4 discusses protection of the CRE occupants and their ability to control the reactor from the control room or from the remote shutdown panels in the event of a smoke challenge.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The worst case single active failure of a component of the CREF, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CREF satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

In the event of a postulated accident, one iodine filtration train is required to provide an adequate supply of filtered air to the CRE. To ensure that this requirement is met, both CREF trains must be OPERABLE. The basis for this approach is that two trains are required to satisfy all design requirements (i.e., one train is needed to mitigate the event and other train is assumed to have a single active failure). The failure of both iodine filtration trains could result in exceeding a dose of 5 rem whole body or its equivalent to any part of the body 5 rem TEDE in the event of a large radioactive release.

Each CREF train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CREF train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Prefilters, HEPA filters, and carbon adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CREF trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for postulated accidents, and that CRE occupants are protected from smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design conditions, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be procedurelized, and consist of

.

BASES

LCO (continued)

stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CREF trains must be OPERABLE to ensure that the CRE will remain habitable during and following a postulated accident (i.e., LOCA, main steam line break, rod ejection, and fuel handling accident).

In MODE 5 or 6, the CREF is also required to cope with a failure of the Gaseous Waste Processing System.

ACTIONS

A.1

With one CREF train inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the OPERABLE CREF train is adequate to perform the CRE occupant protection function. However, the overall system reliability is reduced. The 7 day Completion Time is based on the low probability of a postulated accident occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1, B.2, and B.3

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of postulated accident consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body 5 rem TEDE), or inadequate protection of CRE occupants from smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a postulated accident, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of postulated accident

BASES

ACTIONS (continued)

consequences, and that CRE occupants are protected from smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a postulated accident occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a postulated accident. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, if any Required Action and Completion Time of Condition A or B cannot be met, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner.

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREF train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREF train in the emergency mode. This action ensures that the other train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

BASES

ACTIONS (continued)

E.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREF trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the CRE. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

F.1

With both Iodine Filtration trains and associated Air Conditioning trains inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary (i.e., Condition B), the CREF may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations which dry out any moisture accumulated in the carbon from humidity in the ambient air should be performed. Each Iodine filtration train must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy.

SR 3.7.10.2

This SR verifies that the required CREF train testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, carbon adsorber efficiency, minimum flow rate, and the physical properties of the activated carbon. Specific test Frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

This SR verifies that each CREF train starts and operates on an actual or simulated actuation signal. The Frequency of 24 months is based on industry operating experience and is consistent with the typical refueling cycle.

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of postulated accident consequences is no more than 5 rem whole body or its equivalent to any part of the body 5 rem TEDE and the CRE occupants are protected from smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of postulated accident consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Mitigating actions, or compensatory measures, are discussed in Regulatory Guide 1.196, Section 2.7.3, (Ref. 5) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating measures as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures (Ref. 7). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis postulated accident consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

BASES

REFERENCES

1. FSAR Section 9.4.
2. Chapter 15.
3. FSAR Section 6.4.
4. FSAR Section 9.5.
5. Regulatory Guide 1.196.
6. NEI 99-03, "Control Room Habitability Assessment," March 2003.
7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2005, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability" (ADAMS Accession No. ML040300694).

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Air Conditioning System (CRACS)

BASES

BACKGROUND

The CRACS provides temperature control for the control room envelope (CRE) following isolation of the control room.

The CRACS operates in the recycling mode with fresh outside air makeup. There are four normal system 75% capacity identical fresh air intake trains. For each intake train, the fresh air is taken from outside through motorized damper, electric heater, and prefilter. The fresh filtered air is then mixed with the CRE recycled air. The mixed air is then directed through one of the four 75% capacity associated air conditioning train. Each air conditioning train consists of volume control manual damper, cooling coil, moisture separator, fan suction and discharge silencers, supply air fan, HEPA filter, steam humidifier, non-return damper, volume control electric damper, and fire dampers. The conditioned air is supplied to the control room envelope (CRE) areas. Electric heaters are installed in the supply air ducts to maintain individual room temperatures and relative humidity. The exhaust air from the control room envelope (CRE) areas is directed through the recirculation air shaft and then recycled through the air conditioning trains upstream of the cooling coils for each train. The exhaust air from the CRE can also be recycled through the CREF Iodine Filtration trains if contamination is detected in the CRE. The exhaust from kitchen and sanitary areas is separated from the recycle return air and processed separately.

Two out of four 75% CRACS Air Conditioning trains operating in the recirculation mode with fresh outside makeup air will provide the required temperature in the Main Control Room (MCR) between 65°F to 75°F, and humidity 40% to 60%.

The CRACS operation in maintaining the CRE temperature is discussed in FSAR Section 9.4.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CRACS is to maintain the CRE for 30 days of continuous occupancy.

There are four CRACS trains with two trains normally in operation. During emergency operation, one train is assumed to be out for maintenance and a second train is assumed lost to single failure. The two OPERABLE CRACS trains maintain the MCR temperature between 65°F to 75°F. Redundant detectors and controls are provided for

BASES

APPLICABLE SAFETY ANALYSES (continued)

control room temperature control. The CRACS is designed in accordance with Seismic Category I requirements. The CRACS is capable of removing sensible and latent heat loads from the CRE, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

The CRACS satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Four independent and redundant trains of the CRACS are required to be OPERABLE to ensure that at least two are available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CRACS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in all three trains. These components include the heating and cooling coils, moisture separators, humidifiers, and associated temperature control instrumentation. In addition, the CRACS must be operable to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies, the CRACS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

ACTIONS

A.1

With one or two CRACS train(s) inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRACS trains are adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CRACS train could result in loss of CRACS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or non-safety related cooling means are available.

BASES

ACTIONS (continued)

B.1 and B.2

If any Required Action and Associated Completion Time of Condition A is not met in MODE 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

In MODE 5 or 6, or during movement of irradiated fuel, if the inoperable CRACS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRACS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with three or more CRACS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

If three or more CRACS trains are inoperable in MODE 1, 2, 3, or 4, the CRACS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.11.1 This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the in the control room heat load calculation. This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the CRACS is slow and is not expected over this time period.
REFERENCES	1. FSAR Section 9.4.1.

B 3.7 PLANT SYSTEMS

B 3.7.12 Safeguard Building Controlled Area Ventilation System (SBVS)

BASES

BACKGROUND

The SBVS provides a protected environment in the hot mechanical areas of Safeguards Building Divisions 1, 2, 3, and 4 and the fuel building. The SBVS also filters airborne radioactive particulates from the areas of the active Emergency Core Cooling System (ECCS) components during a Loss of Coolant Accident (LOCA).

The conditioned air supply to all four Safeguard Building Divisions is provided independently for each division by the Electrical Division of Safeguard Ventilation System (Ref. 1). The SBVS supplies the conditioned air for ventilation through a volume control damper and two isolation dampers for each division to the hot mechanical areas of the four Safeguard Building Divisions. The SBVS air supply and exhaust flows are designed to prevent spread of airborne contamination and to maintain a negative pressure in the safeguard buildings and fuel building.

Under normal plant operation, the operational air exhaust from each hot area is drawn independently through a volume control damper and two isolation dampers located on the operational exhaust duct system for each safeguard building. The main exhaust duct of each division is connected to a common concrete duct which runs inside the annulus. The operational air exhaust is then drawn through a concrete duct cell for processing by the normal filtration train of the Nuclear Auxiliary Building Ventilation System prior to release through the plant stack (Ref. 2).

During conditions in which a release of airborne contamination from any of the four hot mechanical areas occurs, the SBVS will redirect the accident air exhaust independently via four separate exhaust lines which join into one common leak-tight exhaust duct inside the annulus. The exhaust duct then connects to an accident exhaust filtration train located in the fuel building. There are two 100% capacity accident iodine exhaust filtration trains in parallel configuration. Each train consists of inlet motor controlled damper, electric heater, pre-filter, upstream HEPA filter, iodine filter with activated carbon, downstream HEPA filter, outlet motor controlled damper, exhaust fan, and non-return damper. The accident air exhaust is processed through one or both independent iodine filtration trains prior to release through the plant stack. The downstream HEPA filter is not credited in the analysis, but serves to collect carbon particles and provides a backup in case the upstream HEPA filter bank fails. The pre-filters remove any large particles in the air and any entrained water droplets present to prevent excessive loading of the HEPA filters and carbon adsorbers.

BASES

BACKGROUND (continued)

In case of a LOCA with assumed ECCS leakage, the accident air exhaust from the safeguard buildings and fuel building is also directed through the accident iodine exhaust filtration trains prior to release through the plant stack.

The SBVS accident iodine filtration train is a standby system which may also be operated during normal plant operations. Upon receipt of an actuating signal, the normal air exhaust from the buildings is isolated and the accident air is redirected through the iodine filtration train.

The SBVS is discussed in FSAR Section 9.4.5 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The SBVS design basis is established by the consequences of the limiting postulated accident, which is a LOCA with assumed ECCS leakage. The analysis of a LOCA, given in Reference 4, assumes ECCS leakage to the safeguard buildings and fuel building is a conservative four gallons a minute. The SBVS consists of two 100% capacity iodine filtration trains in parallel configuration. There are only two iodine filtration trains since only slow failure modes are assumed and filtration efficiency is checked periodically. Both sets of iodine filtration trains are required to be OPERABLE. One SBVS train is then assumed to be lost due to a single failure. The postulated accident analysis assumes that two trains of the SBVS are OPERABLE. The accident analysis accounts for the reduction in airborne radioactive material provided by the one train of this filtration system. The amount of fission products available for release from the safeguard buildings and fuel building is determined for a LOCA. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 5).

The SBVS is not credited in the Fuel Handling Accident evaluation.

The SBVS satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Two independent and redundant trains of SBVS Accident Exhaust Filtration are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power.

The failure of both trains could result in the atmospheric release from the safeguard buildings and fuel building exceeding the 10 CFR 50.34 (Ref. 6) limits in the event of a LOCA.

BASES

LCO (continued)

The SBVS Accident Exhaust Filtration train is considered OPERABLE when it's associated:

a. Fan is OPERABLE;

b. Prefilter, HEPA filter and carbon adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

c. Heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The LCO is modified by a Note allowing the safeguard building and fuel building boundaries to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for safeguard building or fuel building isolation is indicated.

APPLICABILITY

In MODE 1, 2, 3, or 4, the SBVS Accident Exhaust Filtration train is required to be OPERABLE to provide fission product removal associated with the leakage inside the hot areas of the Safeguard Buildings.

In MODE 5 or 6, the SBVS Accident Exhaust Filtration train is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one SBVS Accident Exhaust Filtration train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the SBVS function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable SBVS train, and the remaining SBVS train providing the required protection.

BASES

ACTIONS (continued)

B.1

If the safeguard buildings or fuel building boundary is inoperable in MODE 1, 2, 3, or 4, the SBVS trains may not be able to perform their intended functions. Actions must be taken to restore an OPERABLE safeguard buildings and fuel building boundaries within 24 hours. During the period that the safeguard buildings or fuel building boundary is inoperable, appropriate compensatory measures consistent with the intent, as applicable, of GDC 19 and 10 CFR Part 100 shall be utilized to protect plant personnel from potential hazards such as radioactive contamination,smoke, temperature and relative humidity, and physical security. Preplanned measures shall be available and implemented upon entry into the condition to address these concerns regardless of whether the entry is intentional or unintentional entry. The 24 hour Completion Time is reasonable based on the low probability of a postulated accident occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the safeguard buildings or fuel building boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the associated Completion Time, or when both SBVS Accident Exhaust Filtration trains are inoperable for reasons other than an inoperable safeguard building or fuel building boundary (i.e., Condition B), the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Verifying that safeguards building and fuel building negative pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to safeguards building and fuel building pressure variations and pressure instrument drift during the applicable MODES.

SR 3.7.12.2

Maintaining safeguards building and fuel building OPERABILITY requires verifying each access opening door is closed. However, all safeguards building and fuel building access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.7.12.3

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Monthly heater operation dries out any moisture accumulated in the carbon from humidity in the ambient air. Systems with heaters must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.12.4

This SR verifies that the required SBVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, carbon adsorber efficiency, minimum system flow rate, and the physical properties of the activated carbon (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.5

This SR verifies that each SBVS train starts and operates on an actual or simulated actuation signal. The 24 month Frequency is consistent with Reference 7.

SR 3.7.12.6 and 3.7.12.7

The SBVS exhausts the safeguards building and fuel building atmosphere to the environment through appropriate treatment equipment. Each safety SBVS train is designed to draw down the safeguards building and fuel building to a negative pressure ≥ 0.25 inches of water gauge (wg) in ≤ 305 seconds and maintain the safeguards building and fuel building at a negative pressure ≥ 0.25 inches wg at a flow rate ≤ 2,400 cfm from the safeguards building and fuel building. To ensure that all fission products released to the safeguards building and fuel building are treated, SR 3.7.12.6 and SR 3.7.12.7 verify that a pressure in the safeguards building and fuel building that is less than the lowest postulated pressure external to the safeguards building and fuel building boundaries can be established and maintained. When the SBVS is operating as designed, the establishment and maintenance of safeguards building and fuel building pressure cannot be accomplished if the safeguards building or fuel building boundaries is not intact. Establishment of this pressure is confirmed by SR 3.7.12.6. SR 3.7.12.7 demonstrates that the safeguards building and fuel building can be maintained at a negative pressure ≥ 0.25 inches wg. The primary purpose of these SRs is to ensure safeguards building and fuel building boundary integrity. The secondary purpose of these SRs is to ensure that the SBVS train being tested functions as designed. There is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the SBVS. These SRs need not be performed with each safety SBVS train. The SBVS train used for these Surveillances is staggered to ensure that in addition to the requirements of LCO 3.7.12, either safety SBVS train will perform this test. The inoperability of the SBVS does not necessarily constitute a failure of these Surveillances relative to the safeguards building and fuel building OPERABILITY. Operating experience has shown the safeguards building and fuel building boundaries usually pass these Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

REFERENCES	1. FSAR Section 9.4.6.
	2. FSAR Section 9.4.3.
	3. FSAR Section 9.4.5.
	4. FSAR Section 15.0.
	5. Regulatory Guide 1.25.
	6. 10 CFR 50.34.
	7. Regulatory Guide 1.52, Rev. 3.

B 3.7 PLANT SYSTEMS

B 3.7.13 Safeguards Building Ventilation System Electrical Division (SBVSED)

BASES

BACKGROUND

The SBVSED provides temperature control for the electrical and instrumentation and control rooms of each safeguards building.

The SBVSED consists of four independent trains that provide cooling and heating of the electrical equipment areas of each safeguards building. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for temperature control. The SBVSED can be operated with or without recycled air depending on the outside air temperature.

The SBVSED is an emergency system which also operates during normal unit operations and accident conditions to provide ventilation and cooling in the electrical equipment areas of the safeguards buildings.

Following a loss of offsite power, previously running SBVSED trains return to operation once the emergency diesel generator is started and the associated AC electrical power division is re-energized.

The SBVSED operation in maintaining the safeguards building temperature is discussed in FSAR Section 9.4.6 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the SBVSED is to provide ventilation and air conditioning to the electrical equipment area of the safeguards buildings following any Design Basis Accident (DBA). There are four SBVSED trains, with one train normally in operation in each of the four safeguards buildings. During emergency operation, one train is assumed to be lost to single failure of a diesel generator. The three OPERABLE SBVSED trains maintain their respective safeguards building in a pre-determined temperature range. The SBVSED is designed in accordance with Seismic Category I requirements. The SBVSED is capable of removing sensible and latent heat loads from the safeguards building, which include consideration of equipment heat loads, to ensure equipment OPERABILITY.

The SBVSED satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Four independent trains of the SBVSED are required to be OPERABLE and in operation to ensure that at least three are available, assuming a single failure disabling one train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

BASES

LCO (continued)

The SBVSED is considered to be OPERABLE when the individual components necessary to maintain the safeguards building temperature are OPERABLE and in operation in all four trains. These components include the cooling coils and associated temperature control instrumentation.

APPLICABILITY

In MODES 1, 2, 3, 4, the SBVSED must be OPERABLE and in operation to ensure that the safeguards building electrical equipment areas will not exceed equipment operational requirements following a DBA.

In MODES 5 and 6, the OPERABILITY requirements of the SBVSED is determined by the systems that it supports.

ACTIONS

A.1

With one SBVSED train inoperable or not in operation, action must be taken to restore OPERABLE status within 72 hours. In this condition, the remaining OPERABLE SBVSED trains are adequate to maintain the three remaining safeguards building temperature within limits. A non-safety maintenance train is available to provide temperature control in the affected safeguards building electrical area. However, the overall reliability is reduced because a loss of offsite power would result in loss of SBVSED function in the affected train. The 72 hour Completion Time is based on the low probability of an event occurring, the consideration that the remaining safeguards trains can provide the required safety function, and that alternate, non-safety related cooling means are available.

B.1 and B.2

If any Required Action and associated Completion Time of Condition A is not met, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

Each SBVSED train is verified to be in operation at a frequency of 24 hours to verify that ventilation and air conditioning to the electrical equipment area of each safeguard building. The 24 hour Frequency is appropriate since the train is normally in operation and other indications are available to alert the control room to a failure of a SBVSED train.

SR 3.7.13.2

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the in the safeguards building heat load calculation. This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the SBVSED is slow and is not expected over this time period.

SR 3.7.13.3

This SR verifies proper automatic operation of the SBVSED train on an actual or simulated actuation signal. The SBVSED System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 9.4.6.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the spent fuel storage pool design is given in FSAR Section 9.1.2 (Ref. 1). A description of the Fuel Pool Cooling and Purification System is given in FSAR Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in FSAR Section 15.7.4 (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the spent fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is well within the limits of Table 6 of Regulatory Guide 1.183 (Ref. 5). According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop. The fuel storage pool water level satisfies Criteria 2 of 10 CFR 50.36(d)(2)(ii).
LCO	The spent fuel storage pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

BASES

APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool, since the potential for a release of fission products exists.
ACTIONS	A.1 Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel storage pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	SR 3.7.14.1 This SR verifies sufficient spent fuel storage pool water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience. During refueling operations, the level in the spent fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.
REFERENCES	1. FSAR Section 9.1.2. 2. FSAR Section 9.1.3. 3. FSAR Section 15.7.4. 4. Regulatory Guide 1.25, March 1972. 5. Regulatory Guide 1.183, Table 6, July 2000.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Storage Pool Boron Concentration

BASES

BACKGROUND	The water in the spent fuel storage pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel in the spent fuel storage racks, the design basis for preventing criticality outside the reactor is that there is a 95 percent probability at a 95 percent confidence level, without soluble boron, that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 0.995, including uncertainties and tolerances. The NRC guidelines specify a limiting k_{eff} of 1.0 for normal storage in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains a subcritical condition for the allowed loading patterns. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single independent accident need be considered at one time. For example, the only accident scenario that has a potential for more than negligible positive reactivity effect is an inadvertent misplacement of a new fuel assembly. This accident has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the location of each assembly in accordance with LCO 3.7.16, "Spent Fuel Pool Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.
APPLICABLE SAFETY ANALYSES	Although credit for the soluble boron normally present in the spent fuel pool water is permitted under abnormal or accident conditions, most abnormal or accident conditions will not result in exceeding the limiting reactivity even in the absence of soluble boron. The effects on reactivity of credible abnormal and accident conditions due to temperature increase, boiling, assembly dropped on top of a rack, lateral rack module movement and misplacement of a spent fuel assembly have been analyzed. The spent fuel pool k_{eff} storage limit of 0.95 is maintained during these events by a minimum boron concentration of 500 ppm with boric acid enriched to ≥ 37% B^{10} established by criticality analysis (Ref. 2). Compliance with the LCO minimum boron concentration limit of 500 ppm with boric acid enriched to ≥ 37% B^{10} ensures that the credited concentration is always available. The concentration and enrichment of dissolved boron in the spent fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO	The spent fuel storage pool boron concentration is required to be ≥ 500 ppm boron enriched to ≥ 37% B^{10}. The specified concentration and enrichment of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 2. This concentration of dissolved boron is the minimum required concentration and enrichment for fuel assembly storage and movement within the spent fuel pool.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.
ACTIONS	<u>A.1, A.2.1, and A.2.2</u> When the concentration or enrichment of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration or enrichment of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the spent fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration and enrichment of boron must be restored. This does not preclude movement of a fuel assembly to a safe position. The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR and SR 3.7.15.2 are met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

SR 3.7.15.2

Verification every 24 months that the B^{10} enrichment is within limit ensures that the B^{10} concentration assumed in the accident analyses is available. Since the boron in the spent fuel pool is not exposed to a significant neutron flux, 24 months is considered conservative.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2).

2. UN-TR-08-001, “Spent and New Fuel Storage Analyses for U.S. EPR Topical Report,” March 2008.

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Storage

BASES

BACKGROUND

The high density spent fuel storage racks are divided into two separate and distinct regions as shown in Figure 4.3-1. Region 1, with a maximum of 360 storage locations, is designed to accommodate new fuel assemblies with a maximum enrichment of 5.0 weight percent U-235, or spent fuel assemblies regardless of the combination of initial enrichment and burnup. Region 2, with a maximum of 1000 storage locations, is designed to accommodate spent fuel assemblies in all locations which comply with the combination of initial enrichment and burnup limits specified in Figure 3.7.16-1, Fuel Assembly Burnup Requirements for Region 2.

The water in the spent fuel storage pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel in the spent fuel racks, the design basis for preventing criticality outside the reactor is that there is a 95 percent probability at a 95 percent confidence level, without soluble boron, that the effective multiplication faction (k_{eff}) of the fuel assembly array will be less than 0.995, including uncertainties and tolerances. The NRC guidelines specify a limiting k_{eff} of 1.0 for normal storage in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains a subcritical condition for the allowed loading patterns.

The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal and accident conditions, since only a single independent accident need be considered at one time. For example, the only accident scenario that has the potential for more than negligible positive reactivity effect is an inadvertent misplacement of a new fuel assembly. This accident has is a potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, enriched boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the combination of initial enrichment and burnup of the stored fuel in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.

BASES

APPLICABLE SAFETY ANALYSES	The hypothetical accidents can only take place during or as a result of the movement of an assembly (Refs. 2 and 3). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.15, "Spent Fuel Pool Boron Concentration") prevents criticality. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO. The configuration of fuel assemblies in the spent fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).
LCO	The restrictions on the placement of fuel assemblies within Region 2 of the spent fuel pool in the accompanying LCO, ensure the k_{eff} of the spent fuel storage pool will always remain < 0.995, assuming the pool to be flooded with unborated water and < 0.95, with a boron concentration of greater than 500 ppm and boron enrichment ≥ 37%. Storage of spent fuel is permitted in all Region 2 locations provided that the spent fuel meets the combination of initial enrichment and burnup requirements shown in Figure 3.7.16-1, Fuel Assembly Burnup Requirements for Region 2.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in Region 2 of the spent fuel storage pool.
ACTIONS	A.1 When the requirements of the LCO are not met, action must be immediately initiated to move the non-complying fuel assembly to an acceptable storage location (i.e., Region 1). Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.7.16.1</u> This SR verifies by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1 prior to storing the fuel assembly in Region 2.
REFERENCES	1. Double contingency principle ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2). 2. UN-TR-08-001, "Spent and New Fuel Storage Analyses for U.S. EPR Topical Report," March 2008. 3. U.S. EPR FSAR Section 15.0.3.10, "Fuel Handling Accident."

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit bounds the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.12, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.15, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 50.34 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in FSAR Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates.

BASES

APPLICABLE SAFETY ANALYSES (continued)

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and Main Steam Relief Trains (MSRTs). The Emergency Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Low Head Safety Injection System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and MSRTs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a postulated accident to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a postulated accident.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS	<u>A.1 and A.2</u> DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.
SURVEILLANCE REQUIREMENTS	<u>SR 3.7.17.1</u> This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.
REFERENCES	1. 10 CFR 50.34. 2. FSAR Chapter 15.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate), and the onsite standby power sources (four emergency diesel generators (EDGs)). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E AC Distribution System is divided into redundant load groups (divisions) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each division has connections to two preferred offsite power sources and a dedicated EDG.

Offsite power is supplied to the unit switchyard(s) from the transmission network by at least two transmission lines. Two 100% capacity emergency auxiliary transformers receive power from the switchyard, reduce the voltage to 6.9 kV, and feed the Emergency Power Supply System (EPSS), which provides Class 1E power to the four divisions of safety related plant loads. Three normal auxiliary transformers feed the Normal Power Supply System (NPSS), which provides power to non-safety related plant loads. The Class 1E EPSS is totally independent of the non-Class 1E NPSS. A detailed description of the offsite power network and the circuits to the Class 1E ESF buses is found in FSAR Chapter 8 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus(es).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System. Within 1 minute after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the Protection System. The onsite standby power source for each 6.9 kV ESF bus is a dedicated EDG. EDGs 30XKA10, 30XKA20, 30XKA30, and 30XKA40 are dedicated to ESF Divisions 1, 2, 3, and 4, respectively. An EDG starts automatically on Degraded Grid Voltage or Loss of Offsite Power signals. After the EDG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage, independent of, or coincident with, a Safety Injection System (SIS) Actuation signal. The EDGs will also start and operate in the standby mode without tying to the ESF bus on an SIS Actuation signal alone. Following the trip of offsite power, the

BASES

BACKGROUND (continued)

Protection System strips nonpermanent loads from the ESF bus. When the EDG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the Protection System. The sequencing logic prevents overloading the EDG.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the EDGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a postulated accident, such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the EDG in the process. Within 1 minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

The four divisions of the onsite Class 1E AC Distribution System are arranged into "divisional pairs" to support maintenance activities (i.e., Divisions 1 and 2 are one divisional pair and Divisions 3 and 4 are the other divisional pair). Each division can be aligned to power a subset of loads ("alternate fed loads") in the other division in its divisional pair by means of an "alternate feed". An alternate feed provides a standby source of power to required safety systems, safety support systems, or components that do not have the required redundant trains to support maintenance. The EDGs and busses have been sized to accommodate the alternate fed loads. The alternate feeds are manually controlled and interlocked such that two power sources can not be supplied to any alternate fed bus at the same time.

Ratings for the four EDGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each EDG is 9500 kW with 10% overload permissible for up to 2 hours in any 24 hour period. The ESF loads that are powered from the 6.9 kV ESF buses are listed in Reference 2.

APPLICABLE SAFETY ANALYSES

The initial conditions of postulated accident and anticipated operational occurrences (AOOs) in FSAR Chapter 6 (Ref. 4) and FSAR Chapter 15 (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least two divisions OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC power; and

b. A worst case single failure.

There are four EDGs. Each supports one independent ESF division. Each ESF division provides redundancy for most four division safety systems. However, there are some required safety systems, safety support systems, or components that do have the necessary redundancy to support maintenance. For these systems, safety support systems, or components, an alternate feed is provided between the Class 1E electrical divisional pairs (i.e., between Divisions 1 and 2 and between Divisions 3 and 4) to facilitate maintenance on a division while still providing normal and emergency power to the required loads.

The design of the onsite Class 1E AC Electrical Power Distribution System ensures that, with one EDG or portion of one division's electrical distribution system not available and the associated alternate feed aligned, all safety functions required for a postulated accident, coincident with a single failure and the loss off offsite power will be powered.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power System and separate and independent EDGs for each division ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated accident.

Qualified offsite circuits are those that are described in FSAR Chapter 8 and are part of the licensing basis for the unit.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses.

Each offsite circuit is capable of supplying power to all four divisions. However, normal plant lineup is such that each offsite circuit powers two divisions. Offsite circuit #1 is powered from the switchyard through Emergency Auxiliary Transformer 30BDT01 and feeds Division 1 bus 31BDA and Division 3 bus 33BDA via normal feeder breakers. The circuit can also be aligned to feed Division 2 bus 32BDA and Division 4 bus

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LCO (continued)

34BDA via normally open feeder breakers. Offsite circuit #2 is powered from the switchyard through Emergency Auxiliary Transformer 30BDT02 and feeds Division 2 bus 32BDA and Division 4 bus 34BDA via normal feeder breakers. The circuit can also be aligned to feed Division 1 bus 31BDA and Division 3 bus 33BDA via normally open feeder breakers.

Each EDG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of loss of bus voltage or degraded bus voltage. This will be accomplished within 15 seconds. Each EDG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as EDG in standby with the engine hot and EDG in standby with the engine at ambient conditions. Additional EDG capabilities must be demonstrated to meet required Surveillances, (e.g., capability of the EDG to revert to standby status on a SIS Actuation signal while operating in parallel test mode).

Proper sequencing of loads is a required function for EDG OPERABILITY.

The AC sources in one division must be separate and independent (to the extent possible) of the AC sources in the other divisions. For the EDGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical. A circuit may be connected to more than one ESF bus, with fast transfer capability to the other circuit OPERABLE, and not violate separation criteria. A circuit that is not connected to an ESF bus is required to have OPERABLE fast transfer interlock mechanisms to at least two ESF buses to support OPERABILITY of that circuit.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated accident.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources - Shutdown."

BASES

ACTIONS

A Note prohibits the application of LCO 3.0.4.b when two or more EDGs are inoperable. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with inoperable EDGs and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition D for two offsite circuits inoperable, is entered.

A.2

Required Action A.2, which only applies if the division cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated EDG will not result in a complete loss of safety function of critical redundant required features.

These features are powered from the redundant AC electrical power divisions. Required features are those features required to be OPERABLE by their associated LCOs.

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The division has no offsite power supplying its loads; and

b. A required feature on another division is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

BASES

ACTIONS (continued)

Discovering no offsite power to one division of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other divisions that have offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a postulated accident occurring during this period.

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a postulated accident occurring during this period.

B.1

The Required Actions have been modified by a Note. The Note recognizes that the alternate feed from an OPERABLE EDG in a divisional pair would not provide power if both EDGs in the divisional pair were inoperable. Therefore, this Required Action does not need to be performed.

If one EDG is inoperable and the alternate feed is not aligned, certain required safety systems, safety support systems, and components that do not have 100% four division redundancy do not have sufficient AC power source availability to ensure the completion of all safety functions for a postulated accident coincident with a single failure and the loss of offsite power.

BASES

ACTIONS (continued)

To ensure a highly reliable power source remains with an inoperable EDG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that the EDG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions. Redundant feature failures consist of inoperable features associated with a division, redundant to the division that has an inoperable EDG.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable EDG exists, and

b. A required feature on another division is inoperable.

If at any time during the existence of this Condition (one EDG inoperable), a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one EDG inoperable coincident with one or more inoperable support or supported features, or both, that are associated with the OPERABLE EDGs, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE EDGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY

BASES

ACTIONS (continued)

of the redundant counterpart to the inoperable feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a postulated accident occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE EDGs. If it can be determined that the cause of the inoperable EDG does not exist on the OPERABLE EDGs, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other EDGs, the other EDGs would be declared inoperable upon discovery. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.3.1 is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDGs, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of those EDGs.

In the event the inoperable EDG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE EDGs are not affected by the same problem as the inoperable EDG.

B.4 and B.5

If one EDG is inoperable and the alternate feed is not aligned, certain required safety systems, safety support systems, and components that do not have adequate redundancy to support maintenance, do not have sufficient AC power source availability to ensure the completion of all safety functions for a postulated accident coincident with a single failure and the loss of offsite power.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition B without the alignment of the alternate feed for a period that should not exceed 72 hours. With the alternate feed aligned, the electric power sources required by GDC 17 are available at the required voltage and capacity for the nuclear station and capable of withstanding a system

BASES

ACTIONS (continued)

contingency such as (a) a single failure involving loss of generation by the nuclear unit, any other critical generation source, or loss of power from a transmission system element, or (b) a double failure involving a loss of power from the transmission network and the loss of one division of onsite AC power.

In Condition B, the remaining OPERABLE EDGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 72 hour Completion Time to align the alternate feed from a division containing an OPERABLE EDG takes into account the capacity and capability of the remaining AC sources and the low probability of a postulated accident occurring during this period.

The 120 day Completion Time to restore an EDG to OPERABLE with the alternate feed aligned in its divisional pair is reasonable since its operation is not assumed in the safety analysis to mitigate the consequences of postulated accidents or AOOs, it provides a reasonable time for repairs, and the low probability of a postulated accident or AOO occurring during this period.

C.1 and C.2

The Required Actions have been modified by a Note. The Note recognizes that the alternate feed from an OPERABLE EDG in a divisional pair would not provide power if both EDGs in the divisional pair were inoperable. Therefore, this Required Action does not need to be performed.

With one EDG in both divisional pairs inoperable and the alternate feeds not aligned, there may be no remaining standby AC sources for certain required safety systems, safety support systems, and components that do not have 100% four division redundancy. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system may be the only source of AC power for certain required safety systems, support systems, or components at this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power for certain required safety systems, safety support systems, and components that do not have 100% four division redundancy, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize

BASES

ACTIONS (continued)

the risk associated with this level of degradation. According to Reference 6, operation may continue for a period that should not exceed 2 hours. Also in accordance with Reference 6, operation may continue with one EDG in both divisional pairs inoperable and one alternate feed aligned for a period that should not exceed 72 hours.

If two EDGs in one divisional pair are inoperable, required safety systems, safety support systems, and components do not have sufficient AC power source availability to ensure the completion of all safety functions for a postulated accident coincident with a single failure and the loss of offsite power. According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 72 hours. Completing Required Action C.1 restores the required redundancy in AC power source for required safety systems, safety support systems, and components necessary to ensure completion of the safety function. If two EDGs in one divisional pair are inoperable, the remaining OPERABLE EDG divisional pair and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a postulated accident occurring during this period.

D.1 and D.2

Required Action D.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one circuit without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two offsite circuits inoperable, based upon the assumption that two complete safety divisions are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety related divisions.

The Completion Time for Required Action D.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

BASES

ACTIONS (continued)

a. All offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition D (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system may not have the capability to effect a safe shutdown and to mitigate the effects of a postulated accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more EDGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a postulated accident or AOO. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

BASES

ACTIONS (continued)

E.1 and E.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in deenergization. Therefore, the Required Actions of Condition E are modified by a Note to indicate that when Condition E is entered with no AC source to any division, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems - Operating," must be immediately entered. This allows Condition E to provide requirements for the loss of one offsite circuit and one EDG, without regard to whether a division is deenergized. LCO 3.8.9 provides the appropriate restrictions for a deenergized division. According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition E for a period that should not exceed 12 hours.

In Condition E, individual redundancy is lost in the offsite electrical power system and the onsite AC electrical power system is degraded. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition D (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a postulated accident occurring during this period.

F.1

With three or more EDGs inoperable, there may be no remaining standby AC sources for certain required safety systems, safety support systems, and components that do not have 100% four division redundancy. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system may be the only source of AC power for certain required safety systems, support systems, or components at this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power for certain required safety systems, safety support systems, and components that do not have 100% four division redundancy, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation. According to Reference 6, operation may continue for a period that should not exceed 2 hours.

BASES

ACTIONS (continued)

G.1 and G.2

If any Required Action and associated Completion Time of Conditions A, B, C, D, E, or F cannot be met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1

Condition H corresponds to a level of degradation in which all redundancy in the AC electrical power supplies may have been lost. At this severely degraded level, any further losses in the AC electrical power system may cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the EDGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3) and Regulatory Guide 1.137 (Ref. 9), as addressed in FSAR Section 1.9.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of 6210 V is 90% of the nominal 6.9 kV output voltage. This value, which is specified in ANSI C84-1, allows for voltage drop to the terminals of 6600 V motors whose minimum operating voltage is specified as 90% or 5940 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The specified maximum steady state output voltage of 7260 V is equal to the maximum operating voltage specified for 6600 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 6600 V motors is no more than the maximum rated operating voltages.

The specified minimum and maximum frequencies of the EDG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate postulated accidents and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 1 for SR 3.8.1.2 and the Note for SR 3.8.1.7) to indicate that all EDG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the EDGs are started from standby conditions. Standby conditions for an EDG mean that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

In order to reduce stress and wear on diesel engines, some manufacturers recommend a modified start in which the starting speed of EDGs is limited, warmup is limited to this lower speed, and the EDGs are gradually accelerated to synchronous speed prior to loading. These start procedures are the intent of SR 3.8.1.2 Note 2, which is only applicable when such modified start procedures are recommended by the manufacturer.

SR 3.8.1.7 requires that, at a 184 day Frequency, the EDG starts from standby conditions and achieves required voltage and frequency within 15 seconds. The 15 second start requirement supports the assumptions of the design basis LOCA analysis in FSAR Chapter 15 (Ref. 5).

The 15 second start requirement is not applicable to SR 3.8.1.2 (see Note 2) when a modified start procedure as described above is used. If a modified start is not used, the 15 second start requirement of SR 3.8.1.7 applies.

SURVEILLANCE REQUIREMENTS (continued)

Since SR 3.8.1.7 requires a 15 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2.

In addition to the SR requirements, the time for the EDG to reach steady state operation, unless the modified EDG start method is employed, is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The 31 day Frequency for SR 3.8.1.2 is consistent with Regulatory Guide 1.9 (Ref. 3). The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of EDG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This Surveillance verifies that the EDGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the EDG is connected to the offsite source.

Although no power factor requirements are established by this SR, the EDG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the EDG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

The 31 day Frequency for this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 3).

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one EDG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful EDG start must precede this test to credit satisfactory performance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour of EDG operation at full load plus 10%.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and facility operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during EDG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 9). This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

The 92 day Frequency is appropriate considering the reliability and redundancies of the system.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Transfer of each 6.9 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The 24 month Frequency of the Surveillance is based on engineering judgment and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.8.1.9

Each EDG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the EDG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. For this unit, the single largest load for each EDG is the component cooling water pump with a horsepower (HP) rating of 1250 HP. This Surveillance may be accomplished by:

a. Tripping the EDG output breaker with the EDG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the EDG solely supplying the bus.

As required by Regulatory Guide 1.9 (Ref. 3), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between nominal speed and the overspeed trip setpoint, or 115% of nominal speed, whichever is lower.

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The 3 seconds specified is

BASES

SURVEILLANCE REQUIREMENTS (continued)

equal to 60% of a typical 5 second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the EDG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover following load rejection. The 24 month Frequency is consistent with the recommendation of Regulatory Guide 1.9 (Ref. 3).

This SR is modified by a Note. The Note ensures that the EDG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ 0.9. This power factor is representative of the actual inductive loading an EDG would see under postulated accident conditions. Under certain conditions, however, the Note allows the Surveillance to be conducted at a power factor other than ≤ 0.9. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ 0.9 results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.9 while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the EDG excitation levels needed to obtain a power factor of 0.9 may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the EDG. In such cases, the power factor shall be maintained as close as practicable to 0.9 without exceeding the EDG excitation limits.

SR 3.8.1.10

This Surveillance demonstrates the EDG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The EDG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine and generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the EDG experiences following a full load rejection and verifies that the EDG does not trip upon loss of the load. These acceptance criteria provide for EDG damage protection. While the EDG is not expected to experience this transient during an event and continues to be available, this response ensures that the EDG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

The 24 month Frequency is consistent with the recommendation of Regulatory Guide 1.9 (Ref. 3) and is intended to be consistent with expected fuel cycle lengths.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR has been modified by a Note. The Note ensures that the EDG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ 0.9. This power factor is representative of the actual inductive loading an EDG would see under postulated accident conditions. Under certain conditions, however, the Note allows the Surveillance to be conducted at a power factor other than ≤ 0.9. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ 0.9 results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.9 while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the EDG excitation levels needed to obtain a power factor of 0.9 may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the EDG. In such cases, the power factor shall be maintained as close as practicable to 0.9 without exceeding the EDG excitation limits.

SR 3.8.1.11

As required by Regulatory Guide 1.9 (Ref. 3), Section 2.2.5, this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the EDG. It further demonstrates the capability of the EDG to automatically achieve the required voltage and frequency within the specified time.

The EDG autostart time of 15 seconds is derived from requirements of the accident analysis to respond to a postulated large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability is achieved.

The requirement to verify the connection and power supply of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the EDG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, the Low Head Safety Injection valves are not desired to be stroked open, or residual heat removal (RHR) systems performing a decay heat removal function are not desired to be realigned. In lieu of actual demonstration of connection and loading of loads, testing that

BASES

SURVEILLANCE REQUIREMENTS (continued)

adequately shows the capability of the EDG systems to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 24 months is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1, takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

<u>SR 3.8.1.12</u>

This Surveillance demonstrates that the EDG automatically starts and achieves the required voltage and frequency within the specified time (15 seconds) from an actual or simulated Safety Injection System actuation signal and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.12.d ensures that permanently connected loads are energized from the offsite electrical power system on a SIS actuation without loss of offsite power.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The requirement to verify the connection of permanent loads is intended to satisfactorily show the relationship of these loads to the offsite power loading logic.

The Frequency of 24 months is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is to minimize wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

SR 3.8.1.13

This Surveillance demonstrates that EDG noncritical protective functions are bypassed on an actual or simulated Loss of Offsite Power signal on the emergency bus concurrent with an actual or simulated SIS actuation signal. Noncritical automatic trips are all automatic trips except:

a. Engine overspeed,

b. Generator differential current,

c. Low lube oil pressure,

d. High jacket water temperature, and

e. Low Essential Service Water pressure.

The noncritical trips are bypassed during postulated accidents and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The EDG availability to mitigate the postulated accident is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the EDG.

The 24 month Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.14

Regulatory Guide 1.9 (Ref. 3) requires demonstration once per fuel cycle that the EDGs can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 2 hours of which is at a load equivalent to 105% - 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the EDG. The EDG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

The load band is provided to avoid routine overloading of the EDG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

The 24 month Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1, and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. Note 2 ensures that the EDG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ 0.9. This power factor is representative of the actual inductive loading an EDG would see under postulated accident conditions. Under certain conditions, however, Note 2 allows the Surveillance to be conducted as a power factor other than ≤ 0.9. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ 0.9 results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.9 while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the EDG excitation levels needed to obtain a power factor of 0.9 may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the EDG. In such cases, the power factor shall be maintained close as practicable to 0.9 without exceeding the EDG excitation limits.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 15 seconds. The 15 second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. The 24 month Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3).

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the EDG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY. The requirement that the diesel has operated for at least 2 hours at full load conditions prior to performance of this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 3). Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all EDG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.16

As required by Regulatory Guide 1.9 (Ref. 3), Table 1, this Surveillance ensures that the manual synchronization and automatic load transfer from the EDG to the offsite source can be made and the EDG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart logic is reset to allow the EDG to reload if a subsequent loss of offsite power occurs. The EDG is considered to be in ready to load status when the EDG is at rated speed and voltage, the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequence timing is reset.

The Frequency of 24 months is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1, and takes into consideration unit conditions required to perform the Surveillance.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODES 1, 2, 3 or 4 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following

BASES

SURVEILLANCE REQUIREMENTS (continued)

corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.17

Demonstration of the test mode override ensures that the EDG availability under accident conditions will not be compromised as the result of testing and the EDG will automatically reset to ready to load operation if a SIS actuation signal is received during operation in the test mode. Ready to load operation is defined as the EDG running at rated speed and voltage with the EDG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 10), paragraph 5.2.4.6(b).

SR 3.8.1.17b ensures that emergency loads are energized from the offsite electrical power system on a SIS actuation without a loss of offsite power. SR 3.8.1.17b also demonstrates that the emergency loading was not affected by EDG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable.

This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The 24 month Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1, takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance could perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODES 1, 2, 3, or 4 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing

BASES

SURVEILLANCE REQUIREMENTS (continued)

OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes.

These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.18

In the event of a postulated accident coincident with a loss of offsite power, the EDGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the EDG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with a SIS actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the EDG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 24 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 24 months.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for EDGs. The reason for Note 2 is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODES 1, 2, 3 or 4 is further

BASES

SURVEILLANCE REQUIREMENTS (continued)

amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes.

These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODES 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.19

This Surveillance demonstrates that the EDG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the EDGs are started simultaneously.

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1.

This SR is modified by a Note. The reason for the Note is to minimize wear on the EDG during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR Chapter 8.
3. Regulatory Guide 1.9, Rev. 4.
4. FSAR Chapter 6.
5. FSAR Chapter 15.
6. Regulatory Guide 1.93, Rev. 0, December 1974.

BASES

REFERENCES (continued)

7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.

8. 10 CFR 50, Appendix A, GDC 18.

9. Regulatory Guide 1.137, Rev. 1, October 1979.

10. IEEE Standard 308-2001.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, AC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving handling irradiated fuel.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many postulated accidents that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from accident analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration;

BASES

BACKGROUND (continued)

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both;

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems; and

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite emergency diesel generator (EDG) power.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. Two OPERABLE Emergency Diesel Generators (EDGs) in one divisional pair are required to be OPERABLE by LCO 3.8.10, to ensure a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and EDGs ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es). Qualified offsite circuits are those that are described in FSAR Chapter 8 and are part of the licensing basis for the unit.

Each offsite circuit is capable of supplying power to all four divisions. However, normal plant lineup is such that each offsite circuit powers two divisions. Offsite circuit #1 is powered from the switchyard through Emergency Auxiliary Transformer 30BDT01 and feeds Division 1 bus 31BDA and Division 3 bus 33BDA via normal feeder breakers. The circuit can also be aligned to feed Division 2 bus 32BDA and Division 4 bus 34BDA via normally open feeder breakers. Offsite circuit #2 is powered from the switchyard through Emergency Auxiliary Transformer 30BDT02 and feeds Division 2 bus 32BDA and Division 4 bus 34BDA via normal

BASES

LCO (continued)

feeder breakers. The circuit can also be aligned to feed Division 1 bus 31BDA and Division 3 bus 33BDA via normally open feeder breakers.

Each EDG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This sequence must be accomplished within 15 seconds. The EDG must be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as EDG in standby with the engine hot and EDG in standby at ambient conditions.

It is acceptable for divisions to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required divisions.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

BASES

ACTIONS (continued)

A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF division. Although two divisions are required by LCO 3.8.10, the one division with offsite power available may be capable of supporting sufficient required features to allow continuation of irradiated fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3, B.1, B.2, and B.3

With the offsite circuit not available to all required divisions, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With one or both required EDG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive Moderator Temperature Coefficient (MTC) must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC source and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

BASES

ACTIONS (continued)

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a division is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a de-energized division.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.12 and SR 3.8.1.18 are not required to be met because the Safety Injection System actuation signal is not required to be OPERABLE. SR 3.8.1.17 is not required to be met because the required OPERABLE EDG(s) is not required to undergo periods of being synchronized to the offsite circuit. SR 3.8.1.20 is excepted because starting independence is not required with the DG(s) that is not required to be operable.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE EDG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 6.9 kV ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the EDG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the EDG and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES

None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

BASES

BACKGROUND

Each emergency diesel generator (EDG) is provided with a storage tank having a fuel oil capacity sufficient to operate that diesel for a period of 3 1/2 days while the EDG is supplying maximum post loss of coolant accident load demand discussed in FSAR Section 9.5.4.2 (Ref. 1). The maximum load demand is calculated using the assumption that a minimum of any two EDGs are available. This onsite fuel oil capacity is sufficient to operate the EDGs for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from storage tank to day tank by either of two transfer pumps associated with each storage tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve or tank to result in the loss of more than one EDG.

For proper operation of the standby EDGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

The EDG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated EDG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting a minimum of 3 1/2 days of operation. The onsite storage in addition to the engine oil sump is sufficient to ensure 7 days of continuous operation. This supply is sufficient to allow the operator to replenish lube oil from outside sources.

Each EDG has an air start system with adequate capacity for five successive start attempts on the EDG without recharging the air start receiver(s).

APPLICABLE SAFETY ANALYSIS

The initial conditions of postulated accident and anticipated operational occurrences (AOO) analyses in FSAR Chapter 6 (Ref. 4), and in FSAR Chapter 15 (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The EDGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are

BASES

APPLCABLE SAFETY ANALYSIS (continued)

discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, and the air start subsystem support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Stored diesel fuel oil is required to have sufficient supply for 3 1/2 days of full load operation. It is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate at full load for 3 1/2 days. This requirement, in conjunction with an ability to obtain replacement supplies within 3 1/2 days, supports the availability of EDGs required to shut down the reactor and to maintain it in a safe condition for an AOO or a postulated accident with loss of offsite power. EDG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown."

The starting air system is required to have a minimum capacity for five successive EDG start attempts without recharging the air start receivers.

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated accident. Since stored diesel fuel oil, lube oil, and the starting air subsystem support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits when the associated EDG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each EDG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable EDG subsystem. Complying with the Required Actions for one inoperable EDG subsystem may allow for continued operation, and subsequent inoperable EDG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

BASES

ACTIONS (continued)

A.1

In this Condition, the 3 1/2 day fuel oil supply for an EDG is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 3 day supply. These circumstances may be caused by events, such as full load operation required after an inadvertent start while at minimum required level, or feed and bleed operations, which may be necessitated by increasing particulate levels or any number of other oil quality degradations. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the EDG inoperable. This period is acceptable based on the remaining capacity (> 3 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

With lube oil inventory < 750 gallons, sufficient lubricating oil to support 7 days of continuous EDG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the EDG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.3. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated EDG inoperable. The 7 day Completion Time allows for further evaluation, resampling and re-analysis of the EDG fuel oil.

BASES

ACTIONS (continued)

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if an EDG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the EDG would still be capable of performing its intended function.

E.1

With starting air receiver pressure < 435 psig, sufficient capacity for five successive EDG start attempts does not exist. However, as long as the receiver pressure is > 220 psig, there is adequate capacity for at least one start attempt, and the EDG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the EDG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most EDG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With a Required Action and associated Completion Time not met, or one or more EDG's fuel oil, lube oil, or starting air subsystem not within limits for reasons other than addressed by Conditions A through E, the associated EDG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each EDG's operation for 3 1/2 days at full load. The 3 1/2 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an onsite or offsite location.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.2

This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of full load operation for each EDG. The 750 gallon requirement is based on the EDG manufacturer consumption values for the run time of the EDG. Implicit in this SR is the requirement to verify the capability to transfer the lube oil from its storage location to the EDG, when the EDG lube oil sump does not hold adequate inventory for 7 days of full load operation without the level reaching the manufacturer recommended minimum level.

A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since EDG starts and run time are closely monitored by the unit staff.

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057-R2000 (Ref. 6),

b. Verify in accordance with the tests specified in ASTM D975-2006 (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60°F of ≥ 27° and ≤ 39° when tested in accordance with ASTM D1298-1999 R2005 (Ref. 6), a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point of ≥ 125°F, and

BASES

SURVEILLANCE REQUIREMENTS (continued)

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-2004 E2005 (Ref. 6) or a water and sediment content within limits when tested in accordance with ASTM D2709-1996 R2006 (Ref. 6).

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-2006 (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975-2006 (Ref. 7), except that the analysis for sulfur may be performed in accordance with ASTM D1552-2003, ASTM D2622-2005, or ASTM D4294-2003 (Ref. 6). The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on EDG operation. This Surveillance ensures the availability of high quality fuel oil for the EDGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D5452-2005 (Ref. 6). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each EDG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. A start cycle is defined by the EDG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 92 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during EDG operation. Water may come from any of several sources, including condensation, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

REFERENCES

1. FSAR Section 9.5.4.2.
2. Regulatory Guide 1.137.
3. ANSI N195-1976, Appendix B.
4. FSAR Chapter 6.
5. FSAR Chapter 15.
6. ASTM Standards: D4057-1995 R2000; D975-2006; D1298-1999 R2005; D4176-2004 E2005; D2709-1996 R2006; D1552-2003; D2622-2005; D4294-2003; D5452-2005.
7. ASTM Standards, D975-2006, Table 1.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC vital bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 250 VDC electrical power system consists of four independent and redundant safety related Class 1E DC electrical power divisions. Each division consists of one 250 VDC battery, two 100% capacity battery chargers, and all the associated control equipment and interconnecting cabling.

One battery charger in each division is capable of being alternately fed from the other division in the divisional pair. The alternate feeds are provided between Divisions 1 and 2 and between Divisions 3 and 4. The alternate feeds are controlled such that the requirements of independence and redundancy between divisional pairs are maintained.

During normal operation, the 250 VDC system load is powered from one of the battery chargers with the battery floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

The DC electrical power subsystems provide the control power for their associated Class 1E AC power load group, 6.9 kV switchgear, and 480 V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the AC vital buses.

The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution System - Operating," and LCO 3.8.10, "Distribution Systems - Shutdown."

Each 250 VDC battery is separately housed in a ventilated room apart from its charger and distribution center. Each subsystem is located in an area separated physically and electrically from the other subsystems to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E divisions for the batteries, inverters, or distribution panels. One battery charger in each division can be powered from the other division in the

BASES

BACKGROUND (continued)

divisional pair. The chargers are interlocked such that only one charger can be connected to the DC subsystem at one time.

Each battery has adequate storage capacity to meet the duty cycle(s) discussed in FSAR Chapter 8 (Ref 4). The battery is designed with additional capacity above that required by the design duty cycle to allow for temperature variations and other factors.

The batteries for the DC subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit is 210 V.

The battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. This specific gravity corresponds to an open circuit battery voltage of approximately 250 V for a 120 cell battery (i.e., cell voltage of 2.065 volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. Once fully charged with its open circuit voltage ≥ 2.065 Vpc, the battery cell will maintain its capacity for 30 days without further charging per manufacturer's instructions. Optimal long term performance however, is obtained by maintaining a float voltage 2.20 to 2.25 Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge. The nominal float voltage of 2.22 Vpc corresponds to a total float voltage output of 266.4 V for a 120 cell battery as discussed in FSAR Chapter 8 (Ref. 4).

Each DC subsystem battery charger has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient excess capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads discussed in FSAR Chapter 8 (Ref. 4).

The battery charger is normally in the float-charge mode. Float-charge is the condition in which the charger is supplying the connected loads and the battery cells are receiving adequate current to optimally charge the battery. This assures the internal losses of a battery are overcome and the battery is maintained in a fully charged state.

When desired, the charger can be placed in the equalize mode. The equalize mode is at a higher voltage than the float mode and charging current is correspondingly higher. The battery charger is operated in the equalize mode after a battery discharge or for routine maintenance.

BASES

BACKGROUND (continued)

Following a battery discharge, the battery recharge characteristic accepts current at the current limit of the battery charger (if the discharge was significant, e.g., following a battery service test) until the battery terminal voltage approaches the charger voltage setpoint. Charging current then reduces exponentially during the remainder of the recharge cycle. Lead-calcium batteries have recharge efficiencies of greater than 95%, so once at least 105% of the ampere-hours discharged have been returned, the battery capacity would be restored to the same condition as it was prior to the discharge. This can be monitored by direct observation of the exponentially decaying charging current or by evaluating the amp-hours discharged from the battery and amp-hours returned to the battery.

APPLICABLE SAFETY ANALYSES

The initial conditions of postulated accidents and anticipated operational occurrences (AOOs) analyses in FSAR Chapter 6 (Ref. 5) and FSAR Chapter 15 (Ref. 6), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the EDGs, emergency auxiliaries, and control and switching during all MODES of operation.

DC control and I&C power must be provided to the safety systems, safety support systems, or components that do not have four 100% redundant divisions. If one EDG is out of service, an alternate feed is provided to allow one battery charger in that division to be powered from the other division in the divisional pair to ensure completion of safety functions.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power, and

b. A worst-case single failure.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The DC electrical power subsystems, each subsystem consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the division are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated accident. Loss of any division DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

BASES

LCO (continued)

An OPERABLE DC electrical power subsystem requires all required batteries and one battery charger to be operating and connected to the associated DC bus.

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated accident.

The DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.5, "DC Sources - Shutdown."

ACTIONS

A.1, A.2, and A.3

Condition A represents one division with one required battery charger inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning a required inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 12 hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 12 hours, avoiding a premature shutdown with its own attendant risk.

BASES

ACTIONS (continued)

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 2 hours, and the charger is not operating in the current-limiting mode, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit mode that is necessary during the recovery period following a battery discharge event that the DC system is designed for.

If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 12 hours (Required Action A.2).

Required Action A.2 requires that the battery float current be verified as less than or equal to 2 amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 12 hour period the battery float current is not less than or equal to 2 amps this indicates there may be additional battery problems and the battery must be declared inoperable.

Required Action A.3 limits the restoration time for the required inoperable battery charger to 72 hours. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., use of a spare battery charger). The 72 hour Completion Time reflects a reasonable time to effect restoration of the required qualified battery charger to OPERABLE status.

B.1

Condition B represents one subsystem with one battery inoperable. With the battery inoperable, the DC bus is being supplied by the required OPERABLE battery charger. Any event that results in a loss of the AC bus supporting the battery charger will also result in loss of DC to that division. Recovery of the AC bus, especially if it is due to a loss of offsite power, will be hampered by the fact that many of the components necessary for the recovery (e.g., diesel generator control and field flash, AC load shed and diesel generator output circuit breakers, etc.) likely rely upon the battery. In addition the energization transients of any DC loads

ACTIONS (continued)

that are beyond the capability of the required battery charger and normally require the assistance of the battery will not be able to be brought online. The 2 hour limit allows sufficient time to effect restoration of an inoperable battery given that the majority of the conditions that lead to battery inoperability (e.g., loss of battery charger, battery cell voltage less than 2.07 V, etc.) are identified in Specifications 3.8.4, 3.8.5, and 3.8.6 together with additional specific Completion Times.

C.1

Condition C represents one required division with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected division. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution subsystem.

If one of the required DC subsystems is inoperable for reasons other than Condition A or B (e.g., inoperable battery charger and associated inoperable battery), the remaining DC subsystems have the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst-case single failure could, however, result in the loss of the minimum necessary DC subsystems to mitigate a worst case accident, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 7) and reflects a reasonable time to assess unit status as a function of the inoperable DC subsystem and, if the DC subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

D.1 and D.2

If the inoperable DC subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 7).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the battery chargers, which support the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the minimum float voltage established by the battery manufacturer (2.20 Vpc or 264 V at the battery terminals). This voltage maintains the battery plates in a condition that supports maintaining the grid life (expected to be approximately 20 years). The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 8).

SR 3.8.4.2

This SR verifies the design capacity of the battery chargers. According to Regulatory Guide 1.32 (Ref. 9), the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

This SR provides two options. One option requires that each battery charger be capable of supplying 400 amps at the minimum established float voltage for 8 hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power. The time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least 2 hours.

The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, and the exponential decay in charging current. The battery is recharged when the measured charging current is ≤ 2 amps.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 24 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.8.4.3

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 24 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed during refueling operations, or at some other outage, with intervals between tests not to exceed 24 months.

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1, 2, 3 or 4 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-2001.
4. FSAR Chapter 8.
5. FSAR Chapter 6.
6. FSAR Chapter 15.
7. Regulatory Guide 1.93, December 1974.
8. IEEE-450-2002.
9. Regulatory Guide 1.32, March 2004.
10. Regulatory Guide 1.129, February 2007.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND

A description of the DC/UPS System is provided in the Bases for LCO 3.8.4, "DC Sources – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of postulated accidents and anticipated operational occurrences in FSAR Chapter 6 (Ref. 1) and FSAR Chapter 15 (Ref. 2), assume the Protection System (PS) is OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the EDGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many postulated accidents that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

BASES

BACKGROUND (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case accidents which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, has found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Two DC subsystems, each of which consists of one battery, one required battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the division, are required to be OPERABLE to support two divisions of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The DC subsystems sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

b. Required features needed to mitigate a fuel handling accident are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1, A.2, and A.3

Condition A represents one subsystem with one battery charger inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 12 hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 12 hours.

BASES

ACTIONS (continued)

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 2 hours, and the charger is not operating in the current-limiting mode, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit mode that is necessary during the recovery period following a battery discharge event that the DC system is designed for.

If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 12 hours (Required Action A.2).

Required Action A.2 requires that the battery float current be verified as less than or equal to 2 amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 12 hour period the battery float current is not less than or equal to 2 amps this indicates there may be additional battery problems and the battery must be declared inoperable.

Required Action A.3 limits the restoration time for the inoperable battery charger to 72 hours. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., balance of plant non-Class 1E battery charger). The 72 hour Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

B.1, B.2.1, B.2.2, and B.2.3

By allowing the option to declare required features inoperable with the associated DC subsystem(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of irradiated fuel assemblies and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in

BASES

ACTIONS (continued)

failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive Moderator Temperature Coefficient (MTC) must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by Specification 3.8.4, "DC Sources - Operating". See the corresponding Bases for Specification 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC subsystems from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. FSAR Chapter 6.

2. FSAR Chapter 15.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Parameters

BASES

BACKGROUND	This LCO delineates the limits on battery float current as well as electrolyte temperature, level, and float voltage for the DC batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4, "DC Sources – Operating" and LCO 3.8.5, "DC Sources - Shutdown." In addition to the limitations of this Specification, the licensee controlled program also implements a program specified in Specification 5.5.16 for monitoring various battery parameters that is based on the recommendations of IEEE Standard 450-2002, "IEEE Recommended Practice For Maintenance, Testing, And Replacement Of Vented Lead-Acid Batteries For Stationary Applications" (Ref. 1). The battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. This specific gravity corresponds to an open circuit battery voltage of approximately 250 V for 120 cell battery (i.e., cell voltage of 2.065 volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. Once fully charged with its open circuit voltage ≥ 2.065 Vpc, the battery cell will maintain its capacity for 30 days without further charging per manufacturer's instructions. Optimal long term performance however, is obtained by maintaining a float voltage 2.20 to 2.25 Vpc. This provides adequate over-potential which limits the formation of lead sulfate and self discharge. The nominal float voltage of 2.22 Vpc corresponds to a total float voltage output of 266.4 V for a 120 cell battery as discussed in FSAR Chapter 8 (Ref. 2).
APPLICABLE SAFETY ANALYSES	The initial conditions of postulated accidents and anticipated operational occurrences in FSAR Chapter 6 (Ref. 1) and FSAR Chapter 15 (Ref. 2), assume the Protection System (PS) is OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the EDGs, emergency auxiliaries, Instrumentation and Control, and control and switching during all MODES of operation. The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least two divisions of DC sources OPERABLE during accident conditions, in the event of: a. An assumed loss of all offsite AC power or all onsite AC power; and b. A worst-case single failure. Battery parameters satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO

Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated accident. Battery parameter limits are conservatively established, allowing continued DC electrical system function even with limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the licensee controlled program is conducted as specified in Specification 5.5.16.

APPLICABILITY

The battery parameters are required solely for the support of the associated DC divisions. Therefore, battery parameter limits are only required when the DC division is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

ACTIONS

A Note has been added providing that, for this LCO, separate Condition entry is allowed for each battery. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable battery. Complying with the Required Actions for battery cell parameters allows for restoration and continued operation, and subsequent out of limit battery cell parameters may be governed by separate Condition entry and application of associated Required Actions.

A.1, A.2, and A.3

With one or more cells in the battery < 2.07 V, the battery cell is degraded. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage (SR 3.8.4.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.6.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells < 2.07 V, and continued operation is permitted for a limited period up to 24 hours.

Since the Required Actions only specify "perform," a failure of SR 3.8.4.1 or SR 3.8.6.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If SR 3.8.6.1 is failed then there is not assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

BASES

ACTIONS (continued)

B.1 and B.2

A battery with float current > 2 amps indicates that a partial discharge of the battery capacity has occurred. This may be due to a temporary loss of a battery charger or possibly due to one or more battery cells in a low voltage condition reflecting some loss of capacity. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage. If the terminal voltage is found to be less than the minimum established float voltage there are two possibilities, the battery charger is inoperable or is operating in the current limit mode. Condition A addresses charger inoperability. If the charger is operating in the current limit mode after 2 hours that is an indication that the battery has been substantially discharged and likely cannot perform its required design functions. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 12 hours (Required Action B.2). The battery must therefore be declared inoperable.

If the float voltage is found to be satisfactory but there are one or more battery cells with float voltage less than 2.07 V, the associated "OR" statement in Condition F is applicable and the battery must be declared inoperable immediately. If float voltage is satisfactory and there are no cells less than 2.07 V there is good assurance that, within 12 hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to a temporary loss of the battery charger.

A discharged battery with float voltage (the charger setpoint) across its terminals indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 12 hours, avoiding a premature shutdown with its own attendant risk.

If the condition is due to one or more cells in a low voltage condition but still greater than 2.07 V and float voltage is found to be satisfactory, this is not indication of a substantially discharged battery and 12 hours is a reasonable time prior to declaring the battery inoperable.

BASES

ACTIONS (continued)

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.4.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.4.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

C.1, C.2, and C.3

With one battery with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established.

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. Required Actions C.1 and C.2 address this potential (as well as provisions in Specification 5.5.16, Battery Monitoring and Maintenance Program). They are modified by a Note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates. The Required Action C.2 requirement to verify that there is no leakage by visual inspection and the Specification 5.5.16.b item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from Annex D of IEEE Standard 450-2002. They are performed following the restoration of the electrolyte level to above the top of the plates. Based on the results of the manufacturer's recommended testing the battery may have to be declared inoperable and the affected cell(s) replaced.

D.1

With one battery with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met.

BASES

ACTIONS (continued)

E.1

With one battery in two or more divisions with battery parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the batteries can still perform their required function, given that multiple batteries are involved. With multiple batteries involved this potential could result in a total loss of function on multiple systems that rely upon the batteries. The longer Completion Times specified for battery parameters are therefore not appropriate, and the parameters must be restored to within limits on all but one division within 2 hours.

F.1

With one battery with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, D, or E, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one or more batteries with one or more battery cells float voltage less than 2.07 V and float current greater than 2 amps indicates that the battery capacity may not be sufficient to perform the intended functions. The battery must therefore be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450 (Ref. 1). The 7 day Frequency is consistent with IEEE-450 (Ref. 1).

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. When this float voltage is not maintained the Required Actions of LCO 3.8.4 ACTION A are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of 2 amps is established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.2 and SR 3.8.6.5

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to 270.0 V at the battery terminals, or 2.25 Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than 2.07 Vpc, are addressed in Specification 5.5.16. SRs 3.8.6.2 and 3.8.6.5 require verification that the cell float voltages are equal to or greater than the short term absolute minimum voltage of 2.07 V. The Frequency for cell voltage verification every 31 days for pilot cell and 92 days for each connected cell is consistent with IEEE-450 (Ref. 1).

SR 3.8.6.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability. The Frequency is consistent with IEEE-450 (Ref. 1).

SR 3.8.6.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 65°F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity. The Frequency is consistent with IEEE-450 (Ref. 1).

SR 3.8.6.6

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.6.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.4.3.

BASES

SURVEILLANCE REQUIREMENTS (continued)

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

It may consist of just two rates; for instance the one minute rate for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test must remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 1) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this 80% limit.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 1), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 1).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance,

BASES

SURVEILLANCE REQUIREMENTS (continued)

corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. IEEE-450-2002.
2. FSAR Chapter 8.
3. FSAR Chapter 6.
4. FSAR Chapter 15.
5. IEEE-485-1997.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital buses. The Uninterruptible Power Supply (UPS) loads can be powered from an AC source or from the station battery. The station battery provides an uninterruptible power source for the Instrumentation and Control (I&C) system power, which includes the Protection System (PS) and Emergency Diesel Generator (EDG) starting logic. Specific details on inverters and their operating characteristics are found in FSAR Chapter 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions for postulated accidents and anticipated operational occurrences in FSAR Chapter 6 (Ref. 2) and FSAR Chapter 15 (Ref. 3), assume the PS is OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of normal and emergency power for 480 VAC loads requiring uninterruptible power and the AC/DC converters that provide power to the I&C system, which includes the PS, during all MODES of operation. This ensures that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and

b. A worst case single failure.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The inverters ensure the availability of AC electrical power for the instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated accident. The inverters also supply motive power to certain ESF components (e.g., containment isolation valves).

BASES

LCO (continued)

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the PS I&C is maintained. The four inverters (one per division) ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the 6.9 kV safety buses are de-energized.

OPERABLE inverters require the associated vital bus to be powered by the inverter with output voltage and frequency within tolerances, and power input to the inverter from a 250 VDC station battery. Alternatively, the UPS power supply may be from an AC source as long as the station battery is available as the uninterruptible power supply.

APPLICABILITY

The required inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated accident.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."

ACTIONS

A.1

With an inverter inoperable, its associated AC vital bus becomes inoperable until it is re-energized from its Class 1E voltage regulated bus. For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures that the vital bus is re-energized within 2 hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its voltage regulated bus, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable inverter cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the PS connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. FSAR Chapter 8.
2. FSAR Chapter 6.
3. FSAR Chapter 15.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of postulated accidents and anticipated operational occurrences in FSAR Chapter 6 (Ref. 1) and FSAR Chapter 15 (Ref. 2), assume the Protection System (PS) is OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of required power to the Instrumentation and Control (I&C) system, which includes the PS and Emergency Diesel Generator starting logic so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the inverter to each AC vital bus during MODES 5 and 6 ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many accidents that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case accidents which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, has found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The inverter ensures the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated accident. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 6.9 kV safety buses are de-energized. OPERABILITY of the inverter requires that the AC vital bus be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The inverter required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

BASES

APPLICABILITY (continued)

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1, A.2.1, A.2.2, and A.2.3

If two divisions are required by LCO 3.8.10, "Distribution Systems - Shutdown," the remaining OPERABLE Inverters may be capable of supporting sufficient required features to allow continuation of irradiated fuel movement and operations with a potential for positive reactivity additions. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of irradiated fuel assemblies and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive Moderator Temperature Coefficient (MTC) must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverter and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

BASES

ACTIONS (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a voltage regulated bus.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. FSAR Chapter 6.
2. FSAR Chapter 15.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems - Operating

BASES

BACKGROUND

The onsite Class 1E AC, DC, and AC vital bus electrical power distribution systems are divided by division into four redundant and independent AC, DC, and AC vital bus electrical power distribution subsystems.

The AC electrical power subsystem for each division consists of several Class 1E 6.9 kV buses and secondary 480 buses, distribution panels, motor control centers and load centers. Each 6.9 kV bus has at least two separate and independent offsite sources of power as well as a dedicated onsite emergency diesel generator (EDG) source. Each 6.9 kV bus is normally connected to a preferred offsite source. After a failure of the emergency auxiliary transformer supplying preferred offsite power to a 6.9 kV bus, a high speed bus transfer is used to connect the alternate offsite power source to the 6.9 kV bus. If all offsite sources are unavailable, the onsite EDG supplies power to the 6.9 kV bus. Control power for the 6.9 kV breakers is supplied from the Class 1E batteries. Additional description of this system may be found in the Bases for LCO 3.8.1, "AC Sources - Operating," and the Bases for LCO 3.8.4, "DC Sources - Operating."

The secondary AC electrical power distribution subsystem for each division includes the safety related switchgear, load centers, and motor control centers shown in Table B 3.8.9-1.

The 480 VAC vital buses are arranged with one 480V motor control center per division and are normally powered from the inverters. The alternate power supply for the vital buses are Class 1E voltage regulated buses powered from the same division as the associated inverter, and its use is governed by LCO 3.8.7, Inverters – Operating." Each voltage regulated bus is powered from a Class 1E AC source.

The DC electrical power distribution system consists of 250 VDC buses and distribution panels.

The list of all required DC and AC vital distribution buses is presented in Table B 3.8.9-1.

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of postulated accidents and anticipated operational occurrences in FSAR Chapter 6 (Ref. 1) and FSAR Chapter 15 (Ref. 2), assume the Protection System (PS) is OPERABLE. The AC, DC, and AC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC, DC, and AC vital bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC electrical power; and

b. A worst case single failure.

The distribution systems satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The required power distribution subsystems listed in Table 3.8.9-1 ensure the availability of AC, DC, and AC vital bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated accident. The designated AC, DC, and AC vital electrical power distribution subsystems are required to be OPERABLE.

Maintaining the Divisions 1, 2, 3, and 4 AC, DC, and AC vital bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

OPERABLE AC electrical power distribution subsystems require the associated buses, load centers, motor control centers, and distribution panels to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses and distribution panels to be energized to their proper voltage from the associated battery or charger. OPERABLE vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated inverter via inverted DC voltage, inverter using an AC source, or Class 1E voltage regulated bus.

BASES

LCO (continued)

In addition, tie breakers between redundant safety related AC, DC, and AC vital bus power distribution subsystems, if they exist, must be open unless they are being utilized to power an alternate feed. The alternate feed is interlocked to prevent sources from two divisions supplying a bus at the same time. In addition, interlocks prevent inadvertently paralleling two EDGs together. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, that could cause the failure of a redundant subsystem and a loss of essential safety function(s). This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 6.9 kV buses from being powered from the same offsite circuit.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated accident.

Electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.10, "Distribution Systems - Shutdown."

ACTIONS

A.1

With one or more of the required AC electrical power distribution subsystems inoperable and a loss of function has not occurred, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in one of the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC electrical power distribution subsystem must be restored to OPERABLE status within 8 hours.

Condition A worst scenario is one 6.9 kV AC electrical power distribution subsystem out for maintenance and another in the same divisional pair (Divisions 1 and 2 or Divisions 3 and 4) without AC power (i.e., no offsite power to the division and the associated EDG inoperable). In this

BASES

ACTIONS (continued)

Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining divisions by stabilizing the unit, and on restoring power to the affected division. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected division, to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the remaining divisions with AC power.

Required Action A.1 is modified by a Note that requires the applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," to be entered for DC division made inoperable by inoperable power distribution subsystems. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. Inoperability of a distribution system can result in loss of charging power to batteries and eventual loss of DC power. This Note ensures that the appropriate attention is given to restoring charging power to batteries, if necessary, after loss of distribution systems.

B.1

With one or more AC vital subsystems inoperable, and a loss of function has not yet occurred, the remaining OPERABLE AC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum ESF functions not being supported. Therefore, the required AC vital bus must be restored to OPERABLE status within 2 hours by powering the bus from the associated inverter via inverted DC, inverter using internal AC source, or Class 1E voltage regulated bus.

Condition B represents one or more AC vital buses without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining vital buses and restoring power to the affected vital bus.

BASES

ACTIONS (continued)

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected division; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a postulated accident occurring during this period.

<u>C.1</u>

With one or more DC electrical power distribution subsystems inoperable, and a loss of function has not yet occurred, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in one of the remaining DC electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the DC buses and distribution panels must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery, charger, or inverter.

Condition C represents one or more DC buses or distribution panels without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining divisions and restoring power to the affected division.

BASES

ACTIONS (continued)

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected division; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).

D.1 and D.2

If any Required Action and associated Completion Time of Conditions A, B, or C cannot be met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

Condition E corresponds to a level of degradation in the electrical power distribution system that causes a required safety function to be lost. When more than one inoperable electrical power distribution subsystem results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the required AC, DC, and AC vital electrical power distribution systems are functioning properly with the correct circuit breaker alignment. This includes the alternate feeds, when aligned. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. FSAR Chapter 6.

2. FSAR Chapter 15.

3. Regulatory Guide 1.93, December 1974.

Table B 3.8.9-1 (page 1 of 1)
AC, DC, and AC Vital Electrical Power Distribution Subsystems

VOLTAGE	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
AC ELECTRICAL POWER DISTRIBUTION SUBSYSTEMS				
6.9 kV	Switchgear 31BDA, 31BDB, 31BDC, 31BDD	Switchgear 32BDA, 32BDB, 32BDD	Switchgear 33BDA, 33BDB 33BDD	Switchgear 34BDA, 34BDB, 34BDC, 34BDD
480 V	Load Centers 31BMA, 31BMB, 31BMC, 31BMD Motor Control Centers 31BNA01, 31BNA02, 31BNB01, 31BNB02, 31BNB03, 31BNC01, 31BND01	Load Centers 32BMA, 32BMB, 32BMD Motor Control Centers 32BNA01, 32BNA02, 32BNB01, 32BNB02, 32BNB03, 32BND01	Load Centers 33BMA, 33BMB, 33BMD Motor Control Centers 33BNA01, 33BNA02, 33BNB01, 33BNB02, 33BNB03, 33BND01	Load Centers 34BMA, 34BMB, 34BMC, 34BMD Motor Control Centers 34BNA01, 34BNA02, 34BNB01, 34BNB02, 34BNB03, 34BNC01, 34BND01
AC VITAL ELECTRICAL POWER DISTRIBUTION SUBSYSTEMS				
480 V	31BRA	32BRA	33BRA	34BRA
DC ELECTRICAL POWER DISTRIBUTION SUBSYSTEMS				
250 V	31BUC	32BUC	33BUC	34BUC

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND	A description of the AC, DC, and vital AC electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."
APPLICABLE SAFETY ANALYSES	The initial conditions of postulated accidents and anticipated operational occurrences in FSAR Chapter 6 (Ref. 1) and FSAR Chapter 15 (Ref. 2), assume the Protection System (PS) is OPERABLE. The AC, DC, and AC vital electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. The OPERABILITY of the AC, DC, and AC vital electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY. The OPERABILITY of the minimum AC, DC, and AC vital electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that: a. The unit can be maintained in the shutdown or refueling condition for extended periods; b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident. The AC, DC, and AC vital electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).
LCO	Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components - all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

BASES

LCO (continued)

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC, DC, and AC vital electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

The AC, DC, and AC vital electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

Although redundant required features may require redundant divisions of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem division may be capable of supporting sufficient required features to allow continuation of irradiated fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this

BASES

ACTIONS (continued)

option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of irradiated fuel assemblies and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive Moderator Temperature Coefficient (MTC) must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC, DC, and AC vital electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal (RHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.3 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.4 is provided to direct declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the AC, DC, and AC vital electrical power distribution subsystems are functioning properly, with all the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

BASES

REFERENCES

1. FSAR Chapter 6.
2. FSAR Chapter 15.

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Low Head Safety Injection System (LHSI) and the Medium Head safety Injection (MHSI) systems are either capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity can then flooded with borated water from the in-containment refueling water storage tank through the open reactor vessel by the use of the LHSI pumps.

The mixing action of the LHSI pump in RHR mode and the natural circulation currents due to thermal driving heads in the upper reactor vessel and refueling cavity further mix the added concentrated boric acid with the water in the refueling canal. The LHSI pump in RHR mode is in operation during refueling (see LCO 3.9.4, "Residual Heat removal (RHR) Loops - High Water Level," and LCO 3.9.5, "Residual heat removal (RHR) Loops - Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

BASES

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% Δk/k margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.1, "SHUTDOWN MARGIN (SDM)."

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a k_{eff} ≤ 0.95. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the RCS. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

BASES

ACTIONS

A.1

Continuation of positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving positive reactivity additions must be suspended immediately.

Suspension of positive reactivity additions shall not preclude moving a component to a safe position. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

A.2

In addition to immediately suspending positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once action has been initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, and connected portions of the refueling canal and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity or canal was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. Chapter 15.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND	The source range neutron flux monitors are used during refueling operations and prior to criticality to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the nuclear instrumentation system. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The installed source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers the lower six decades of neutron flux (1E+6 cps). The detectors also provide visual indication in the control room and can provide an audible count rate to alert operators to a possible dilution accident. The nuclear instrumentation is designed in accordance with the criteria presented in Reference 1.
APPLICABLE SAFETY ANALYSES	The source range neutron flux monitors have no safety function and are not assumed to function during any design basis accident or transient analysis. However, the source range neutron flux monitors provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in the Technical Specifications.
LCO	This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication in the control room.
APPLICABILITY	In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels.

BASES

ACTIONS

A.1 and A.2

With one required source range neutron flux monitor inoperable redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, positive reactivity additions including introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration specified in the COLR must be suspended immediately. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1 and B.2

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.3.1 is a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is based on operating experience that demonstrates channel failure is rare.

SR 3.9.2.2

SR 3.9.3.2 is the performance of a CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CALIBRATION. The CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The CALIBRATION also includes verification of the audible count rate and alarm function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
2. Chapter 15.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During movement of recently irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining the containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of Regulatory Guide 1.183, Table 6 (Ref. 1). Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of recently irradiated fuel assemblies within containment, the equipment hatch must be closed and held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operations in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary.
During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

BASES

BACKGROUND (continued)

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident involving recently irradiated fuel during refueling.

The Containment Ventilation System includes two subsystems, the full flow purge system and the partial flow purge system. During MODES 1, 2, 3, and 4, the valves in the full flow purge penetrations are secured in the closed position. The valves in the partial flow purge penetrations can be opened intermittently, but are closed automatically by the Protection System. Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The full flow purge system is used for this purpose, and the valves are closed by manual initiation or a high radiation signal.

The partial flow purge system remains operational in MODE 6, and all four valves are also closed by manual initiation or a high radiation signal.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during recently irradiated fuel movements.

APPLICABLE SAFETY ANALYSES

During movement of recently irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving handling recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to recently irradiated fuel (Ref. 2). Fuel handling accidents, include dropping a single irradiated fuel assembly, or a handling tool or heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 34 hours without containment closure capability ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the guideline values specified in Reference 1.

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for penetrations capable of being closed by an OPERABLE Containment Ventilation System. The OPERABILITY requirements for this LCO ensure that the automatic Containment Ventilation System valve closure times specified can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of recently irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

The following guidelines are included in the assessment of systems removed from service during movement of recently irradiated fuel.

- During fuel handling ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay and to avoid unmonitored releases.

- A single normal or contingency method to promptly close primary or secondary containment penetrations exists. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

BASES

ACTIONS

A.1

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open valves will demonstrate that the valves are not blocked for closing. Also, the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed an OPERABLE Containment Ventilation System signal.

The Surveillance is performed every 7 days during movement of recently irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such this Surveillance ensures that a postulated fuel handling accident involving handling recently irradiated fuel that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each Containment Ventilation System valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month Frequency maintains consistency with other similar Protection System instrumentation and valve testing requirements.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

BASES

REFERENCES	1.	Regulatory Guide 1.183, Table 6, July 2000.
	2.	Chapter 15.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Residual Heat Removal (RHR) Loops - High Water Level

BASES

BACKGROUND	The purpose of the Residual Heat Removal (RHR) System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the LHSI heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the LHSI heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.
APPLICABLE SAFETY ANALYSES	If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit the LHSI pump to be removed from operation for short durations, under the condition that the boron concentration is not diluted. This conditional stopping of the LHSI pump does not result in a challenge to the fission product barrier. The RHR System satisfies Criterion 4 of 10 CFR 50.36(d)(2)(ii).
LCO	Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one LHSI / RHR loop is required to be OPERABLE because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one LHSI / RHR loop must be OPERABLE and in operation to provide: a. Removal of decay heat;

BASES

LCO (continued)

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an LHSI pump, an LHSI heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from operation for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1, "Boron Concentration." Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) Loops - Low Water Level."

BASES

ACTIONS

A.1

If the RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Immediate suspension of positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4, A.5, and A.6.

If no RHR loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with four bolts;

b. One door in each air lock must be closed; and

BASES

ACTIONS (continued)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Ventilation System.

With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The minimum flow rate specified is to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. Chapter 5.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) Loops - Low Water Level

BASES

BACKGROUND

The purpose of the Residual Heat Removal (RHR) System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the LHSI heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the LHSI heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(d)(2)(ii).

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, two RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

BASES

LCO (continued)

An OPERABLE RHR loop consists of an LHSI pump, an LHSI heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.4, "Residual Heat Removal (RHR) Loops - High Water Level."

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.4, "Residual Heat Removal (RHR) Loops – High Water Level," and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Immediate suspension of positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

BASES

ACTIONS (continued)

B.3, B.4, and B.5

If no RHR loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with four bolts;

b. One door in each air lock must be closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Ventilation System.

With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions stated above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The minimum flow rate specified is to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.

SR 3.9.5.2

Verification that the required pump is OPERABLE ensures that an additional LHSI pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. Chapter 5.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Cavity Water Level

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to well within regulatory limits, as provided by the guidance of Table 6 of Regulatory Guide 1.183 (Ref. 3).
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1). The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 34 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 3). Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).
LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.

BASES

APPLICABILITY	LCO 3.9.6 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Spent Fuel Storage Pool Water Level."
ACTIONS	<u>A.1</u> With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.
SURVEILLANCE REQUIREMENTS	<u>SR 3.9.6.1</u> Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2). The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.
REFERENCES	1. Regulatory Guide 1.25, March 1972. 2. Chapter 15. 3. Regulatory Guide 1.183, Table 6, July 2000.