Levy Nuclear Plant, Units 1 & 2

COL Application

Part 4

Technical Specifications

Revision 1

PLANT SPECIFIC TECHNICAL SPECIFICATIONS (PSTS)

The AP1000 Generic Technical Specifications (GTS) and Bases of the referenced DCD (See FSAR Table 1.6-201) Section 16.1 are incorporated by reference into these plant specific technical specifications (PSTS) with the following departures and/or supplements. Section A addresses the completion of the bracketed information from the DCD GTS and Bases in response to COL information item 16.1-1. Section B provides a complete copy of the PSTS and Bases suitable for enclosing with the Combined License.

Section A. PSTS and Bases Differences from the DCD GTS and Bases (LNP COL 16.1-1)

1. All generic bracketed items in the GTS and Bases have been completed. Plant-specific bracketed items are addressed in Section A.2.

2. The remaining bracketed items in the GTS and Bases are plant specific and are addressed as follows. PSTS pages reflecting each PSTS change to the DCD GTS and Bases are provided in the Section B clean copy.

GTS 3.3.1 Specification 3.3.1 (Table 3.3.1-1) contains a Reviewer Note which addresses future confirmation of chosen setpoints.

Remove the reviewer note in the PSTS. There is no replacement language.

Justification:
The reviewer's note information for this specification is deleted because it is not intended to be a part of technical specifications.

GTS 3.3.2 Specification 3.3.2 (Table 3.3.2-1) contains a Reviewer Note which addresses future confirmation of chosen setpoints.

Remove the reviewer note in the PSTS. There is no replacement language.

Justification:
The reviewer's note information for this specification is deleted because it is not intended to be a part of technical specifications.

GTS 3.6.4 Specification 3.6.4 contains a Reviewer Note which indicates when the low pressure limit would not be required to be included on sites with specific site characteristics.

Remove the reviewer note in the PSTS. There is no replacement language.

Justification:
The reviewer's note information for this specification is deleted because it is not intended to be a part of the plant specific technical specifications.

GTS 4.1 The bracketed information in the GTS reads:
[Not applicable to AP1000 Design Certification. Site specific information

to be provided by the COL applicant.]

Replace the bracketed information in the GTS with the following:
The site for the Levy Nuclear Plant (LNP) is located in the southern part of Levy County, Florida, east of U.S. Highway 19/98 (SR 55) and near the cities and towns of Inglis, Yankeetown, and Crystal River.

Justification
Bracketed information is replaced to establish the site location for Levy Nuclear Plant, Units 1 and 2 (LNP) consistent with the site location identified in FSAR Section 2.1.1.1.

GTS 4.1.1 The bracketed information in the GTS reads:
[This information will be provided by the combined license applicant.]

Replace the bracketed information in the GTS with the following:

The Site Boundary is shown in Figure 4.1-1.

The Exclusion Area Boundary is shown in Figure 4.1-2.

Justification
Bracketed information is replaced to establish Site Boundary and Exclusion Area Boundary for LNP consistent with the descriptions identified in FSAR Sections 2.1.1.2 and 2.1.1.3.

GTS 4.1.2 The bracketed information in the GTS reads:
[This information will be provided by the combined license applicant.]

Replace the bracketed information in the GTS with the following:
The LPZ is defined by the 3 mile radius from the site center point as shown in Figure 4.1-1.

Justification
Bracketed information is replaced to establish the LPZ for LNP consistent with the descriptions identified in FSAR Section 2.1.3.5.

GTS 5.1.1 The bracketed information in the GTS reads:
The [Plant Manager] shall be responsible for overall unit operations 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.

Replace the bracketed information in the GTS with the following:
The Plant General Manager shall be responsible for overall unit operations and shall delegate in writing the succession to this responsibility during his absence.
The Plant General Manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

Justification
Position titles are used consistent with the FSAR organization description in Section 13.1.

GTS 5.1.2 The bracketed information in the GTS reads:
The [Shift Supervisor (SS)] shall be responsible for the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

Replace the bracketed information in the GTS with the following:
The Nuclear Shift Manager (NSM) shall be responsible for the control room command function. During any absence of the (NSM) from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the (NSM) from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

Justification
Position titles are used consistent with the FSAR organization description in Section 13.1.

GTS 5.2.1.a The bracketed information in the GTS reads:
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].

Replace the bracketed information in the GTS with the following:
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.

Justification:
Progress Energy has established that these requirements will be documented in the FSAR.

GTS 5.2.1.b The bracketed information in the GTS reads:
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.

Replace the bracketed information in the GTS with the following:
The Plant General 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.

Justification
Position titles are used consistent with the FSAR organization description in Section 13.1.

GTS 5.2.2 The bracketed information in the GTS reads:
REVIEWER'S NOTE – [Determination of the unit staff positions, numbers, and qualifications are the responsibility of the COL applicant. Input provided in WCAP-14694, Revision 0, for the MCR staff and WCAP-14655, Revision 1, for other than the MCR staff will be used in the determination. Each of the following paragraphs may need to be corrected to specify the plant staffing requirements.]

Remove the reviewer's note information in the GTS. There is no replacement language.

Justification:
The reviewer's note information for this specification is deleted because it is not intended to be a part of plant specific technical specifications

GTS 5.2.2 The bracketed information in the GTS reads:
[The unit staff organization shall include the following:
a. A non-licensed operator shall be assigned to each reactor containing fuel and an … b., c., d., e. …Policy Statement on Engineering Expertise on Shift.]

Remove the brackets and adopt the bracketed information in the GTS except that 5.2.2.d is omitted.

Justification
Generic TS bracketed information is applicable and adopted except for GTS 5.2.2.d which is no longer necessary due to revisions to Part 26 since the approval of the GTS. The removal of GTS 5.2.2.d is consistent with TSTF-511 identified by NRC as an appropriate change to implement the revisions to Part 26 (See 73 FR79923, Notice of Availability of Model Safety Evaluation, Model No Significant Hazards Determination, and Model Application for Licensees that Wish to Adopt TSTF-511, Revision 0, "Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance With 10 CFR Part 26").

GTS 5.3 The bracketed information in the GTS reads:
- 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.]

Remove the reviewer's note information in the GTS. There is no replacement language.

Justification:
The reviewer's note information for this specification is deleted because it is not intended to be a part of plant specific technical specifications.

GTS 5.3.1 The bracketed information in the GTS reads:
Each member of the unit staff shall meet or exceed the minimum

qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standards acceptable to the NRC staff]. [The staff not covered by Regulatory Guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff.]

Replace the bracketed information in the GTS with the following:
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 through the first refueling outage, the Regulatory Position C.1.b of Regulatory Guide 1.8, Revision 2, 1987, applies: cold license operator candidates meet the training elements defined in ANSI/ANS 3.1-1993 but are exempt from the experience requirements defined in ANSI/ANS 3.1-1993.

Justification:
There are no unit staff members with required qualifications not covered by those identified in Regulatory Guide 1.8, Revision 3. As such, the second sentence is unnecessary.
Qualification requirements are part of Regulatory Guide 1.8 however Revision 3 does not address cold license operators. Therefore, this exception is included.

GTS 5.6.1 The bracketed information in the GTS reads:
[The initial report shall be submitted by April 30 of the year following the initial criticality.]

Remove the brackets and adopt the bracketed information in the GTS.

Justification
Generic TS bracketed information is applicable and adopted.

GTS 5.6.2 The bracketed information in the GTS reads:
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].

Remove the brackets and adopt the bracketed information in the GTS.

Justification
Generic TS bracketed information is applicable and adopted.

Section B. Complete Copy of PSTS and Bases

A complete copy of the PSTS and Bases is provided in this section. The copy provided herein incorporates the plant specific information to replace GTS and Bases bracketed information as discussed in Section A above.

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3.4 REACTOR COOLANT SYSTEM (continued)

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5.0 ADMINISTRATIVE CONTROLS (continued)

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.
ACTUATION DEVICE TEST	An ACTUATION DEVICE TEST is a test of the actuated equipment. This test may consist of verification of actual operation but shall, at a minimum, consist of a continuity check of the associated actuated devices. The ACTUATION DEVICE TEST shall be conducted such that it provides component overlap with the ACTUATION LOGIC TEST.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST shall be conducted such that it provides component overlap with the ACTUATION DEVICE TEST.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two-section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for OPERABILITY. Calibration of instrument channels 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

1.1 Definitions

CHANNEL CALIBRATION (continued)

devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.

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 derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
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 5.6.5. Plant operation within these parameter limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same committed effective dose equivalent as the quantity and isotopic mixture of I-130, I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Table 2.1 of EPA Federal Guidance Report No. 11, “Limiting Values of Radionuclide

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)

Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, September 1988.

DOSE EQUIVALENT XE-133

DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same effective dose equivalent as the quantity and isotopic mixture of noble gases (Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138) actually present. The dose conversion factors used for this calculation shall be those listed in Table III.1 of EPA Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," EPA 402-R-93-081, September 1993.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE

LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE, such as that from seals or valve packing, 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;

1.1 Definitions

LEAKAGE (continued)

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System (primary to secondary LEAKAGE); or

4. RCS LEAKAGE through the passive residual heat removal heat exchanger (PRHR HX) to the In-containment Refueling Water Storage Tank (IRWST).

b. <u>Unidentified LEAKAGE</u>

All LEAKAGE that is not identified LEAKAGE.

c. <u>Pressure Boundary LEAKAGE</u>

LEAKAGE (except primary to secondary LEAKAGE and PRHR HX tube LEAKAGE) through a nonisolatable fault in a 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).

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 Chapter 14, Initial Test Program;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

1.1 Definitions

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, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, “RCS Pressure and Temperature (P/T) Limits” and LCO 3.4.14, “Low Temperature Overpressure Protection (LTOP) System.”
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3400 MWt.
REACTOR TRIP CHANNEL OPERATIONAL TEST (RTCOT)	A RTCOT shall be the injection of a simulated or actual signal into the RT (Reactor Trip) CHANNEL as close to the sensor as practicable to verify OPERABILITY of the required interlock and/or trip functions. The REACTOR TRIP CHANNEL OPERATIONAL TEST may be performed by means of a series of sequential, overlapping, or total channel steps so that the entire channel is tested from the signal conditioner through the trip logic.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
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 assembly of highest reactivity worth, which is assumed to be fully withdrawn.

1.1 Definitions

SHUTDOWN MARGIN (continued)

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 RCCAs not capable of being fully inserted, the reactivity worth of these assemblies 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, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, 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.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.

Table 1.1-1 (page 1 of 1)
MODES

MODES	TITLE	REACTIVITY CONDITION (K_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	> 420
4	Safe Shutdown[(b)]	< 0.99	NA	$420 \geq T_{avg} > 200$
5	Cold Shutdown[(b)]	< 0.99	NA	≤ 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 to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in Technical Specifications are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meaning.

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 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 indentions 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.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example, the logical connector AND 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

1.3 Completion Times

DESCRIPTION (continued)

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 first inoperability; and

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 ...." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in example 1.3-3 may not be extended.

1.3 Completion Times

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

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 in 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.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One valve inoperable.	A.1 Restore valve 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 valve is declared inoperable, Condition A is entered. If the valve 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 valve is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second valve is declared inoperable while the first valve is still inoperable, Condition A is not re-entered for the second valve. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable valve. 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 valves 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 valves 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.

1.3 Completion Times

EXAMPLES (continued)

On restoring one of the valves to OPERABLE status the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. This Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second valve being inoperable for > 7 days.

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion

1.3 Completion Times

EXAMPLES (continued)

Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

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. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be 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. The separate Completion Time modified by the phrase “from discovery of failure to meet the LCO” is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal “time zero” for beginning the Completion Time “clock.” In this instance, the Completion Time “time zero” is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

1.3 Completion Times

EXAMPLES (continued)

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 5.	6 hours 36 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.

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.

1.3 Completion Times

EXAMPLES (continued)

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 only applicable to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

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 which 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.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR 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

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 hours 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.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 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

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.

1.3 Completion Times

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 Surveillances, 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

1.4 Frequency

DESCRIPTION (continued)

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:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

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 discusses 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.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) 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 12 hours, 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 in not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside the 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.

If the interval 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, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

1.4 Frequency

EXAMPLES (continued)

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 the 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 were 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 were 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 were 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 were not met), SR 3.0.4 would require satisfying the SR.

1.4 Frequency

EXAMPLES (continued)

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

The interval continues, whether or not the unit operation is in MODE 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 were 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 were 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.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-- **- NOTE -** Not required to be met in MODE 3. --	
Verify parameter is within limits.	24 hours

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 were 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 were 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, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop cold leg temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the WRB-2M DNB correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°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 maintained ≤ 2733.5 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and 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 CONDITIONS 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 3.0.2 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 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; and

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 not be made except 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. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or are part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

3.0 LCO Applicability

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 test 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, additional evaluations and limitations may be required in accordance with Specification 5.5.7, “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 3.1.8 allows 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.
LCO 3.0.8	When an LCO is not met and the associated ACTIONS are not met or an associated ACTION is not provided, action shall be initiated within 1 hour to: a. Restore inoperable equipment and b. Monitor Safety System Shutdown Monitoring Trees parameters Exceptions to this Specification are stated in the individual Specifications.

3.0 LCO Applicability

LCO 3.0.8 (continued)

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.8 is not required.

LCO 3.0.8 is only applicable in MODES 5 and 6.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified Conditions in the Applicability of 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 a 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, which ever 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 a LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. 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. SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODE 2 with keff < 1.0,
MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	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 limits.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within ±1% Δk/k of the normalized 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 may 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 ±1% Δk/k of predicted values.	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.

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 $k_{eff} < 1.0$.	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 within upper limit.	Prior to entering MODE 1 after each refueling

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	--- **- NOTES -** 1. Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm. 2. If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.3.2 shall be repeated once per 14 EFPD during the remainder of the fuel cycle. 3. SR 3.1.3.2 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. --- Verify MTC is within lower limit.	Once each cycle

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All Shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

- NOTE -

Not applicable to gray rods during GRCA swap with OPDMS OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1 Verify SDM to be within the limits specified in the COLR. OR A.1.2 Initiate boration to restore SDM within limit. AND A.2 Be in MODE 3.	1 hour 1 hour 6 hours
B. One rod not within alignment limits.	B.1 Restore rod, to within alignment limits.	8 hours with the On-Line Power Distribution Monitoring System (OPDMS) OPERABLE OR 1 hour with the OPDMS inoperable

	OR	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.2.1.1 Verify SDM to be within the limits specified in the COLR.	1 hour
	OR	
	B.2.1.2 Initiate boration to restore SDM within limit.	1 hour
	AND	
	B.2.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND	
	B.2.3 Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	AND	
	------------------------------------ **- NOTE -** Only required to be performed when OPDMS is inoperable. ------------------------------------	
	B.2.4 Perform SR 3.2.1.1 ($F_Q(Z)$ verification) and SR 3.2.1.2 ($F_Q^W(Z)$ verification).	72 hours
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
		------------------------------------ **- NOTE -** Only required to be performed when OPDMS is inoperable. ------------------------------------	
	B.2.5	Perform SR 3.2.2.1 ($F^N_{\Delta H}$ verification).	72 hours
	AND		
	B.2.6	Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. Required Action and associated Completion Time for Condition B not met.	C.1	Be in MODE 3.	6 hours
D. More than one rod not within alignment limit.	D.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	OR		
	D.1.2	Initiate boration to restore required SDM to within limit.	1 hour
	AND		
	D.2	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	--- **- NOTE -** Not applicable to GRCAs. --- Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days
SR 3.1.4.3	--- **- NOTE -** Not applicable to GRCAs. --- Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 2.47 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$, and b. All reactor coolant pumps operating.	Prior to reactor 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 with OPDMS inoperable.

- 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. <u>OR</u> A.1.2 Initiate boration to restore SDM to within limit. <u>AND</u> A.2 Restore shutdown banks 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 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ with OPDMS inoperable.

- NOTE -

This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control Bank 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 control bank(s) to within limits.	2 hours
B. Control bank 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 control bank sequence and overlap to within limits.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2 with $k_{eff} < 1.0$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify the estimated critical control bank position is within limits specified in the COLR.	Within 4 hours prior to achieving criticality
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	12 hours
SR 3.1.6.3	Verify sequence and overlap limits, specified in the COLR, are met for control banks not fully withdrawn from the core.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Bank Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators by using the On-line Power Distribution Monitoring System (OPDMS).	Once per 8 hours
	<u>OR</u>	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. More than one DRPI per group inoperable.	B.1 Place the control rods under manual control.	Immediately
	<u>AND</u>	
	B.2 Monitor and Record RCS T_{avg}.	Once per 1 hour
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.3 Verify the position of the rods with inoperable position indicators indirectly by using the incore detectors. <u>AND</u> B.4 Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	Once per 8 hours 24 hours
C. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1 Verify the position of the rods with inoperable position indicators by using the OPDMS. <u>OR</u> C.2 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours 8 hours
D. One demand position indicator per bank inoperable for one or more banks.	D.1.1 Verify by administrative means all DRPIs for the affected banks are OPERABLE. <u>AND</u> D.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart. <u>OR</u> D.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours Once per 8 hours 8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	Prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of:

LCO 3.1.3 "Moderator Temperature Coefficient,"
LCO 3.1.4 "Rod Group Alignment Limits,"
LCO 3.1.5 "Shutdown Bank Insertion Limit,"
LCO 3.1.6 "Control Bank Insertion Limits," and
LCO 3.4.2 "RCS Minimum Temperature for Criticality"

may be suspended, and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.b, may be reduced to 3 provided:

a. RCS lowest loop average temperature is ≥ 541°F,

b. SDM is within the limits specified in the COLR, and

c. THERMAL POWER is < 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. <u>AND</u> A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and Associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per SR 3.3.1.7.	Prior to initiation of PHYSICS TESTS
SR 3.1.8.2	Verify the RCS lowest loop average temperature is ≥ 541°F.	30 minutes
SR 3.1.8.3	Verify THERMAL POWER is < 5% RTP.	30 minutes
SR 3.1.8.4	Verify SDM is within the limits specified in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 Chemical and Volume Control System (CVS) Demineralized Water Isolation Valves and Makeup Line Isolation Valves

LCO 3.1.9 Two CVS Demineralized Water Isolation Valves and two Makeup Line Isolation Valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CVS demineralized water isolation valve inoperable. <u>OR</u> One makeup line isolation valve inoperable. <u>OR</u> One CVS demineralized water isolation valve and one makeup line isolation valve inoperable.	A.1 Restore two CVS demineralized water isolation valves and two makeup line isolation valves to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition not met. <u>OR</u> Two CVS demineralized water isolation valves inoperable. <u>OR</u> Two makeup line isolation valves inoperable.	B.1 ------------------------------------ **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. ------------------------------------ Isolate the flow path from the demineralized water storage tank to the Reactor Coolant System by use of at least one closed manual or one closed and de-activated automatic valve.	 1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify two CVS demineralized water isolation valves and two makeup line isolation valves are OPERABLE by stroking the valve closed.	In accordance with the Inservice Testing Program

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with On-line Power Distribution Monitoring System (OPDMS) inoperable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------------------------- **- NOTE -** Required Action A.4 shall be completed whenever this Condition is entered. -------------------------------- $F_Q^C(Z)$ not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination
	AND	
	A.2 Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	AND	
	A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	AND	
	A.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------------------------------- **- NOTE -** Required Action B.4 shall be completed whenever this Condition is entered. -------------------------------- $F_Q^W(Z)$ not within limits.	B.1	Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	<u>AND</u>		
	B.2	Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	<u>AND</u>		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	<u>AND</u>		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the maximum allowable power of the AFD limits
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

- NOTES -

1. During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved at which a power distribution map is obtained.

2. If the OPDMS becomes inoperable while in MODE 1 these surveillances must be performed within 31 days of the last verification of OPDMS parameters.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q^C(Z)$ within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified <u>AND</u> 31 EFPD thereafter

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-- **- NOTE -** If $F_Q^W(Z)$ measurements indicate maximum over z $F_Q^C(Z)$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits; or b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate maximum over z $F_Q^C(Z)$ has not increased. --	
	Verify $F_Q^W(Z)$ within limits.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified <u>AND</u> 31 EFPD thereafter

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 On-line Power Distribution Monitoring System (OPDMS) inoperable.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. -------------------------------- **- NOTE –** Required Actions A.2 and A.3 must be completed whenever Condition A is entered. -------------------------------- $F^N_{\Delta H}$ not within limit.	A.1.1	Restore $F^N_{\Delta H}$ to within limit.	4 hours
	<u>OR</u>		
	A.1.2.1	Reduce THERMAL POWER to < 50% RTP.	4 hours
	<u>AND</u>		
	A.1.2.2	Reduce Power Range Neutron Flux – High trip setpoints to ≤ 55% RTP.	72 hours
	<u>AND</u>		
	A.2	Perform SR 3.2.2.1.	24 hours
	<u>AND</u>		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.3	------------------------------------ **- NOTE -** THERMAL POWER does not have to be reduced to comply with this Required Action. ------------------------------------ Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP <u>AND</u> Prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	-- **- NOTE -** If the OPDMS becomes inoperable while in MODE 1 these Surveillances must be performed within 31 days of the last verification of OPDMS parameters. -- Verify $F^N_{\Delta H}$ within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

LCO 3.2.3 The AFD in %-flux-difference units shall be maintained within the limits specified in the COLR.

- NOTE -

The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER $\geq$ 50% RTP and with the On-Line Power Distribution Monitoring System (OPDMS) inoperable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP and with the OPDMS inoperable.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPTR not within limit.	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	<u>AND</u>		
	A.2	Perform SR 3.2.4.1.	Once per 12 hours
	<u>AND</u>		
	A.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>		

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.5 ------------------------------------ **- NOTES -** 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed whenever Required Action A.5 is performed. ------------------------------------ Normalize excore detectors to restore QPTR to within limit.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>	
	A.6 ------------------------------------ **- NOTE -** Perform Required Action A.6 only after Required Action A.5 is completed. ------------------------------------ Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
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	-- **- NOTES -** 1. With one power range channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. -- Verify QPTR within limit by calculation.	7 days
SR 3.2.4.2	-- **- NOTE -** Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER ≥ 75% RTP. -- Verify QPTR is within limit using a minimum of 4 symmetric pairs of fixed incore detectors.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.5 OPDMS – Monitored Parameters

LCO 3.2.5 The following parameters shall not exceed their operating limits as specified in the COLR:

a. Peak kw/ft(Z)
b. $F^N_{\Delta H}$
c. DNBR
d. SDM.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP with OPDMS OPERABLE for parameters a, b, and c.

MODES 1 and 2 and OPDMS OPERABLE for parameter d.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the parameters a. through c. above not within limits.	A.1 Restore all parameters to within limits.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 ------------------------------------ **- NOTE -** If the power distribution parameters are restored to within their limits while power is being reduced, operation may continue at the power level where this occurs. ------------------------------------ Reduce THERMAL POWER to < 50% RTP.	 4 hours
C. Parameter d above not within limits.	C.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.5.1	Verify the parameters a. through d. to be within their limits.	24 hours with OPDMS alarms OPERABLE <u>OR</u> 12 hours with OPDMS alarms inoperable

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function 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 Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s).	Immediately
B. One manual initiation device inoperable.	B.1 Restore manual initiation device to OPERABLE status. <u>OR</u> B.2.1 Be in MODE 3. <u>AND</u> B.2.2 Open reactor trip breakers (RTBs).	48 hours 54 hours 55 hours
C. One manual initiation device inoperable.	C.1 Restore manual initiation device to OPERABLE status. <u>OR</u> C.2 Open RTBs.	48 hours 49 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or two Power Range Neutron Flux – High channels inoperable.	D.1.1 Reduce THERMAL POWER to ≤ 75% RTP.	12 hours
	<u>AND</u>	
	D.1.2 Place one inoperable channel in bypass or trip.	6 hours
	<u>AND</u>	
	D.1.3 With two inoperable channels, place one channel in bypass and one channel in trip.	6 hours
	<u>OR</u>	
	D.2.1 Place inoperable channel(s) in bypass.	6 hours
	<u>AND</u>	
	------------------------------------ **- NOTE -** Only required to be performed when OPDMS is inoperable and the Power Range Neutron Flux input to QPTR is inoperable. ------------------------------------	
	D.2.2 Perform SR 3.2.4.2 (QPTR verification).	Once per 12 hours
	<u>OR</u>	
	D.3 Be in MODE 3.	12 hours
E. One or two channels inoperable.	E.1.1 Place one inoperable channel in bypass or trip.	6 hours
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	E.1.2 With two channels inoperable, place one channel in bypass and one channel in trip.	6 hours
	<u>OR</u>	
	E.2 Be in MODE 3.	12 hours
F. THERMAL POWER between P-6 and P-10, one or two Intermediate Range Neutron Flux channels inoperable.	F.1.1 Place one inoperable channel in bypass or trip.	2 hours
	<u>AND</u>	
	F.1.2 With two channels inoperable, place one channel in bypass and one channel in trip.	2 hours
	<u>OR</u>	
	F.2 Reduce THERMAL POWER to < P-6.	2 hours
	<u>OR</u>	
	F.3 Increase THERMAL POWER to > P-10.	2 hours
G. THERMAL POWER between P-6 and P-10, three Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. THERMAL POWER < P-6, one or two Intermediate Range Neutron Flux channels inoperable.	H.1 Restore three of four channels to OPERABLE status.	Prior to increasing THERMAL POWER to > P-6

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. One or two Source Range Neutron Flux channels inoperable.	I.1	Suspend operations involving positive reactivity additions.	Immediately
J. Three Source Range Neutron Flux channels inoperable.	J.1	Open RTBs.	Immediately
K. One or two channels inoperable.	K.1.1	Place one inoperable channel in bypass or trip.	6 hours
	AND		
	K.1.2	With two channels inoperable, place one channel in bypass and one channel in trip.	6 hours
	OR		
	K.2	Reduce THERMAL POWER to < P-10.	12 hours
L. One or two channels/ divisions inoperable.	L.1	Restore three of four channels/divisions to OPERABLE status.	6 hours
	OR		
	L.2	Be in MODE 3.	12 hours
M. One or two interlock channels inoperable.	M.1	Verify the interlocks are in required state for existing plant conditions.	1 hour
	OR		
	M.2.1	Place the Functions associated with one inoperable interlock channel in bypass or trip.	7 hours
	AND		

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	M.2.2 With two interlock channels inoperable, place the Functions associated with one inoperable interlock channel in bypass and with one inoperable interlock channel in trip.	7 hours
	OR	
	M.3 Be in MODE 3.	13 hours
N. One division inoperable.	N.1 Open RTBs in inoperable division.	8 hours
	OR	
	N.2.1 Be in MODE 3, 4, or 5.	14 hours
	AND	
	N.2.2 Open RTBs.	14 hours
O. Two divisions inoperable.	O.1 Restore three of four divisions to OPERABLE status.	1 hour
	OR	
	O.2.1 Be in MODE 3, 4, or 5.	7 hours
	AND	
	O.2.2 Open RTBs.	7 hours
P. One or two channels/ divisions inoperable.	P.1 Restore three of four channels/divisions to OPERABLE status.	48 hours
	OR	
	P.2 Open RTBs.	49 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Q. One or two Source Range Neutron Flux channel inoperable.	Q.1 Restore three of four channels to OPERABLE status.	48 hours
	<u>OR</u>	
	Q.2 Open RTBs.	49 hours
R. Required Source Range Neutron Flux channel inoperable.	R.1 Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	R.2 Close unborated water source isolation valves.	1 hour
	<u>AND</u>	
	R.3 Perform SR 3.1.1.1.	1 hour <u>AND</u> Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

- NOTE -

Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	-- **- NOTES -** 1. Adjust nuclear instrument channel in the Protection and Safety Monitoring System (PMS) if absolute difference is > 1% RTP. 2. Required to be met within 12 hours after reaching 15% RTP. 3. If the calorimetric heat balance is < 70% RTP, and if the nuclear instrumentation channel indicated power is: a. lower than the calorimetric measurement by > 1%, then adjust the nuclear instrumentation channel upward to match the calorimetric measurement. b. higher than the calorimetric measurement, then no adjustment is required. -- Compare results of calorimetric heat balance to nuclear instrument channel output.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	-- **- NOTES -** 1. Adjust the conversion factor, ΔT°, in the ΔT power calculation ($q_{\Delta T}$) if absolute difference between $q_{\Delta T}$ and the calorimetric measurement is >1% RTP. 2. Required to be met within 12 hours after reaching 50% RTP. 3. If the calorimetric heat balance is < 70% RTP, and if $q_{\Delta T}$ is: a. lower than the calorimetric measurement by > 5%, then adjust ΔT° to match the calorimetric measurement. b. higher than the calorimetric measurement, then no adjustment is required -- Compare results of calorimetric heat balance to the ΔT power calculation ($q_{\Delta T}$) output.	24 hours
SR 3.3.1.4	-- **- NOTES -** 1. Adjust nuclear instrument channel in PMS if absolute difference is ≥ 3% AFD. 2. Required to be met within 24 hours after reaching 20% RTP. -- Compare results of the incore detector measurements to nuclear instrument channel AXIAL FLUX DIFFERENCE.	31 effective full power days (EFPD)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.5	--- **- NOTE -** Required to be met within 24 hours after reaching 50% RTP. --- Calibrate excore channels to agree with incore detector measurements.	92 EFPD
SR 3.3.1.6	--- **- NOTE -** This Surveillance must be performed on both reactor trip breakers associated with a single division. --- Perform TADOT.	92 days on a STAGGERED TEST BASIS
SR 3.3.1.7	--- **- NOTE -** Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. --- Perform RTCOT.	92 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	---------------------------------- **- NOTE -** This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. ---------------------------------- Perform RTCOT.	------------------------ **- NOTE -** Only required when not performed within previous 92 days ------------------------ Prior to reactor startup <u>AND</u> Four hours after reducing power below P-10 for power and intermediate instrumentation <u>AND</u> Four hours after reducing power below P-6 for source range instrumentation <u>AND</u> Every 92 days thereafter
SR 3.3.1.9	---------------------------------- **- NOTE -** This Surveillance shall include verification that the time constants are adjusted to the prescribed values. ---------------------------------- Perform CHANNEL CALIBRATION.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.10	--- **- NOTE -** Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.11	--- **- NOTE -** Verification of setpoint is not required. --- Perform TADOT.	24 months
SR 3.3.1.12	--- **- NOTE -** Neutron detectors are excluded from response time testing. --- Verify RTS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 5)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.11	NA	NA
	3[(a)],4[(a)],5[(a)]	2	C	SR 3.3.1.11	NA	NA
2. Power Range Neutron Flux						
a. High Setpoint	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.12	≤ 109.06% RTP	109% RTP
b. Low Setpoint	1[(b)],2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10 SR 3.3.1.12	≤ 25.06% RTP	25% RTP
3. Power Range Neutron Flux High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.12	≤ 5.06% RTP with time constant ≥ 2 sec	5.0% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1[(b)],2[(c)]	4	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10 SR 3.3.1.12	≤ 25.23% RTP	25% RTP
	2[(d)]	4	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10 SR 3.3.1.12	≤ 25.23% RTP	25% RTP
5. Source Range Neutron Flux High Setpoint	2[(d)]	4	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10 SR 3.3.1.12	≤ 1.01 E5 cps	1.0 E5 cps
	3[(a)],4[(a)],5[(a)]	4	J,R	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.12	≤ 1.01 E5 cps	1.0 E5 cps
	3[(e)],4[(e)],5[(e)]	1	S	SR 3.3.1.1 SR 3.3.1.10	NA	NA

(a) With Reactor Trip Breakers (RTBs) closed and Plant Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(e) With RTBs open. In this condition, Source Range Function does not provide reactor trip but does provide indication.

Table 3.3.1-1 (page 2 of 5)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	Refer to Note 1 Table 3.3.1-1 (Page 5 of 5)	Refer to Note 1 Table 3.3.1-1 (Page 5 of 5)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	Refer to Note 2 Table 3.3.1-1 (Page 5 of 5)	Refer to Note 2 Table 3.3.1-1 (Page 5 of 5)
8. Pressurizer Pressure a. Low Setpoint	1(f)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≥ 1809.9 psig	1810.3 psig
b. High Setpoint	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≤ 2420.7 psig	2420.3 psig
9. Pressurizer Water Level – High 3	1(f)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≤ 71.05%	71%
10. Reactor Coolant Flow – Low	1(f)	4 per hot leg	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≥ 89.96%(i)	90%(i)
11. Reactor Coolant Pump (RCP) Bearing Water Temperature – High	1,2(f)	4 per RCP	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≤ 190.4°F	190°F
12. RCP Speed – Low	1(f)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≥ 90.9%	91%
13. Steam Generator (SG) Narrow Range Water Level – Low	1,2	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≥ 20.95% span	21% span

(f) Above the P-10 (Power Range Neutron Flux) interlock.

(i) 90% of loop specific indicated flow.

Table 3.3.1-1 (page 3 of 5)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
14. Steam Generator (SG) Narrow Range Water Level – High 2	1,2(k)	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≤ 82.05% span	82% span
15. Safeguards Actuation Input from Engineered Safety Feature Actuation System						
a. Manual	1,2	2	B	SR 3.3.1.11	NA	NA
b. Automatic	1,2	4	M	SR 3.3.1.7	NA	NA
16. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2	4	N	SR 3.3.1.7 SR 3.3.1.10	≥ 9.91 E-6% RTP	1E-5% RTP
b. Power Range Neutron Flux, P-10	1,2	4	N	SR 3.3.1.7 SR 3.3.1.10	≥ 9.94% RTP ≤ 10.06% RTP	10% RTP
c. Pressurizer Pressure, P-11	1,2	4	N	SR 3.3.1.7 SR 3.3.1.10	≤ 1970.4 psig	1970 psig
17. Reactor Trip Breakers	1,2 3(j),4(j),5(j)	4 divisions with 2 RTBs per division	O,P	SR 3.3.1.6	NA	NA
18. Reactor Trip Breaker (RTB) Undervoltage and Shunt Trip Mechanisms	1,2 3(j),4(j),5(j)	1 each per RTB mechanism	O,P	SR 3.3.1.6	NA	NA
19. Automatic Trip Logic	1,2 3(j),4(j),5(j)	4 divisions 4 divisions	M Q	SR 3.3.1.7 SR 3.3.1.7	NA NA	NA NA

(k) Above the P-11 (Pressurizer Pressure) interlock.

(j) With Reactor Trip Breakers closed and Plant Control System capable of rod withdrawal.

Table 3.3.1-1 (page 4 of 5)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
20. ADS Stages 1, 2, and 3 Actuation input from engineered safety feature actuation system						
a. Manual	1,2	2 switch sets	B	SR 3.3.1.11	NA	NA
	3(j),4(j),5(j)	2 switch sets	B	SR 3.3.1.11	NA	NA
b. Automatic	1,2	4	M	SR 3.3.1.7	NA	NA
	3(j),4(j),5(j)	4	Q	SR 3.3.1.7	NA	NA
21. Core Makeup Tank Actuation input from engineered safety feature actuation system						
a. Manual	1,2	2 switch sets	B	SR 3.3.1.11	NA	NA
	3(j),4(j),5(j)	2 switch sets	B	SR 3.3.1.11	NA	NA
b. Automatic	1,2	4	M	SR 3.3.1.7	NA	NA
	3(j),4(j),5(j)	4	Q	SR 3.3.1.7	NA	NA

(j) With Reactor Trip Breakers closed and Plant Control System capable of rod withdrawal.

Table 3.3.1-1 (page 5 of 5)
Reactor Trip System Instrumentation

<u>Note 1: Overtemperature ΔT</u>

The ΔT power signal, $q_{\Delta T}$, shall not be less than the measured reactor thermal power by more than 1% of RTP, where the ΔT power signal, $q_{\Delta T}$, is computed as

$q_{\Delta T} = p\ (T_C, P_{PZR})\ [h\ (T_H, P_{PZR}) - h\ (T_C, P_{PZR})]/\Delta T°$

where:

T_C	=	$[(1 + T_1 s)/(1 + T_2 s)]\ T_{COLD}$, where T_{COLD} is the measured cold leg temperature, °F, with lead/lag compensation applied to compensate for cold leg-to-core transit time
T_H	=	$[(1 + T_3 s)/(1 + T_4 s)]\ T_{HOT}$, where T_{HOT} is the measured hot leg temperature, °F, with lead/lag compensation applied to compensate for core-to-hot leg transit time $\tau_1 \geq$ [*] sec $\tau_2 \leq$ [*] sec $\tau_4 \geq$ [*] sec $\tau_5 \leq$ [*] sec
$\rho\ (T_C, P_{PZR})$	=	density of water at the measured cold leg temperature in the cold leg (T_C), °F, and measured pressurizer pressure, $P_{PZR,}$ psia
$h\ (T, P_{PZR})$	=	enthalpy of water at the specified measured temperature (T_H or T_C) and measured pressurizer pressure, P_{PZR}, psia
ΔT°	=	a conversion factor, such that the value of $q_{\Delta T}$ is 100 percent at normal rated thermal power
s	=	Laplace transform operator

And the Overtemperature ΔT setpoint shall not exceed the following nominal Trip Setpoint by more than 0.2% of RTP for T_{HOT}; 0.2% of RTP for T_{COLD}; 0.06% of RTP for pressure; and 0.14% of RTP for ΔI.

$OT\Delta T_{SP} = OT\Delta T^{\circ}_{SP} - f_1(\Delta I)$

where:

$OT\Delta T^{\circ}_{SP}$ = f (P_{PZR}, T_C), determined by interpolation from tables [*] of allowable core thermal power as a function of core inlet temperature at various pressures

P_{PZR} and T_C pressurizer pressure and cold leg temperature, are as defined above

$f_1(\Delta I)$	=	[*] {[*] + ($q_t - q_b$)]	when $q_t - q_b \leq$ -[*]% RTP
		0% of RTP	when -[*]% RTP < $q_t - q_b \leq$ [*]% RTP
		-[*] {($q_t - q_b$) – [*]}	when $q_t - q_b$ > [*]% RTP

where q_t and q_b are percent RTP in the upper and lower halves of the core respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

*These values denoted with [*] are specified in the COLR.

<u>Note 2: Overpower ΔT</u>

The Overpower ΔT setpoint shall not exceed the following nominal Trip Setpoint by more than 0.2% of RTP for T_{HOT}; and 0.2% of RTP for T_{COLD}.

$OP\Delta T_{SP} = K_4 - f_2\ (\Delta I)$

Where:

$K_4 \leq$ [*]

$f_2(\Delta I)$ = [*]

*These values denoted with [*] are specified in the COLR.

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

- NOTES -

1. Separate condition entry is allowed for each Function.

2. The Conditions for each Function are given in Table 3.3.2-1. If the Required Actions and associated Completion Times of the first Condition are not met, refer to the second Condition.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or divisions inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or division(s).	Immediately
B. One or two channels or divisions inoperable.	B.1 Place one inoperable channel or division in bypass or trip.	6 hours
	AND	
	B.2 With two inoperable channels or divisions, place one inoperable channel or division in bypass and one inoperable channel or division in trip.	6 hours
C. One channel inoperable.	C.1 Place inoperable channel in bypass.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required division inoperable.	D.1 Restore required division to OPERABLE status.	6 hours
E. One switch or switch set inoperable.	E.1 Restore switch and switch set to OPERABLE status.	48 hours
F. One channel inoperable.	F.1 Restore channel to OPERABLE status.	72 hours
	<u>OR</u>	
	F.2.1 Verify alternate radiation monitors are OPERABLE.	72 hours
	<u>AND</u>	
	F.2.2 Verify control room isolation and air supply initiation manual controls are OPERABLE.	72 hours
G. One switch, switch set, channel, or division inoperable.	G.1 Restore switch, switch set, channel, and division to OPERABLE status.	72 hours
H. One channel inoperable.	H.1 Place channel in trip.	6 hours
I. One or two channels inoperable.	I.1 Place one inoperable channel in bypass or trip.	6 hours
	<u>AND</u>	
	I.2 With two inoperable channels, place one channel in bypass and one channel in trip.	6 hours
J. One or two interlock channels inoperable.	J.1 Verify the interlocks are in the required state for the existing plant conditions.	1 hour
	<u>OR</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	J.2.1	Place the Functions associated with one inoperable interlock channel in bypass or trip.	7 hours
	<u>AND</u>		
	J.2.2	With two interlock channels inoperable, place the Functions associated with one inoperable interlock channel in bypass and with one inoperable interlock channel in trip.	7 hours
K. Required Action and associated Completion Time not met.	K.1	------------------------------------ **- NOTE -** LCO 3.0.8 is not applicable. ------------------------------------ Suspend movement of irradiated fuel assemblies.	Immediately
L. Required Action and associated Completion Time not met.	L.1	Be in MODE 3.	6 hours
M. Required Action and associated Completion Time not met.	M.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	M.2	Be in MODE 4.	12 hours
N. Required Action and associated Completion Time not met.	N.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	N.2	Be in MODE 4 with the RCS cooling provided by the RNS.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
O. Required Action and associated Completion Time not met.	O.1	Be in MODE 3.	6 hours
	AND		
	O.2	Be in MODE 5.	36 hours
P. Required Action and associated Completion Time not met.	P.1	------------------------------------ **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. ------------------------------------	
		Isolate the affected flow path(s).	24 hours
	AND		
	P.2.1	Isolate the affected flow path(s) by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	7 days
	OR		
	P.2.2	Verify the affected flow path is isolated.	Once per 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
Q. Required Action and associated Completion Time not met.	Q.1	------------------------------------ **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. ------------------------------------ Isolate the affected flow path(s) by use of at least one closed manual or closed and de-activated automatic valve.	6 hours
	OR		
	Q.2.1	Be in MODE 3.	12 hours
	AND		
	Q.2.2	Be in MODE 4.	18 hours
R. Required Action and associated Completion Time not met.	R.1	Be in MODE 3.	6 hours
	AND		
	R.2.1.1	------------------------------------ **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. ------------------------------------ Isolate the affected flow path(s).	12 hours
	AND		
	R.2.1.2	Verify the affected flow path is isolated.	Once per 7 days
	OR		
	R.2.2	Be in MODE 4 with the RCS cooling provided by the RNS.	30 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
S. Required Action and associated Completion Time not met.	S.1 Be in MODE 3. <u>AND</u> S.2.1.1 Be in MODE 4 with the RCS cooling provided by the RNS. <u>AND</u> S.2.1.2 ----------------------------------- **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. ----------------------------------- Isolate the affected flow path(s). <u>AND</u> S.2.1.3 Verify the affected flow path is isolated. <u>OR</u> S.2.2 Be in MODE 5.	6 hours 24 hours 30 hours Once per 7 days 42 hours
T. Required Action and associated Completion Time not met.	T.1.1 ----------------------------------- **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. ----------------------------------- Isolate the affected flow path(s). <u>AND</u>	 6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	T.1.2.1 Isolate the affected flow path(s) by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	7 days
	<u>OR</u>	
	T.1.2.2 Verify the affected flow path is isolated.	Once per 7 days
	<u>OR</u>	
	T.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	T.2.2 Be in MODE 5.	42 hours
U. Required Action and associated Completion Time not met.	U.1 Be in MODE 5.	12 hours
	<u>AND</u>	
	U.2 Initiate action to open the RCS pressure boundary and establish a pressurizer level ≥ 20%.	12 hours
V. Required Action and associated Completion Time not met.	V.1 Restore the inoperable channel(s).	168 hours
	<u>OR</u>	
	V.2.1 Be in MODE 5.	180 hours
	<u>AND</u>	
	V.2.2 Initiate action to open the RCS pressure boundary and establish a pressurizer level ≥ 20%.	180 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
W. Required Action and associated Completion Time not met.	W.1	If in MODE 5 with the RCS open and < 20% pressurizer level, initiate action to be MODE 5 with the RCS pressure boundary open and ≥ 20% pressurizer level.	Immediately
	<u>AND</u>		
	W.2	If in MODE 5, isolate the flow path from the demineralized water storage tank to the RCS by use of at least one closed and de-activated automatic valve or closed manual valve.	Immediately
	<u>AND</u>		
	W.3	If in MODE 6, initiate action to be in MODE 6 with the water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately
	<u>AND</u>		
	W.4	Suspend positive reactivity additions.	Immediately
X. Required Action and associated Completion Time not met.	X.1	If in MODE 5 with RCS open and < 20% pressurizer level, initiate action to be in MODE 5 with RCS open and ≥ 20% pressurizer level.	Immediately
	<u>AND</u>		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	X.2	If in MODE 6 with upper internals in place, initiate action to be in MODE 6 with the upper internals removed.	Immediately
	<u>AND</u>		
	X.3	Suspend positive reactivity additions.	Immediately
Y. Required Action and associated Completion Time not met.	Y.1	Suspend positive reactivity additions.	Immediately
	<u>AND</u>		
	Y.2	If in MODE 4, be in MODE 5.	12 hours
	<u>AND</u>		
	Y.3	If in MODE 4 or 5, initiate action to establish a pressurizer level $\geq$ 20% with the RCS pressure boundary intact.	12 hours
	<u>AND</u>		
	Y.4	If in MODE 6, initiate action to be in MODE 6 with the water level $\geq$ 23 feet above the top of the reactor vessel flange.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
Z. Required Action and associated Completion Time not met.	Z.1	------------------------------------ **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. ------------------------------------	
		Isolate the affected flow path(s) by use of at least one closed manual or closed and deactivated automatic valve.	6 hours
	<u>OR</u>		
	Z.2.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	Z.2.2	Be in MODE 4 with the RCS cooling provided by the RNS.	30 hours
AA. Required Action and associated Completion Time not met.	AA.1.1	------------------------------------ **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. ------------------------------------	
		Isolate the affected flow path(s).	24 hours
	<u>AND</u>		

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	AA.1.2.1 Isolate the affected flow path(s) by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	7 days
	<u>OR</u>	
	AA.1.2.2 Verify the affected flow path is isolated.	Once per 7 days
	<u>OR</u>	
	AA.2.1 If in MODE 4, be in MODE 5.	12 hours
	<u>AND</u>	
	AA.2.2 If in MODE 4 or 5, initiate action to establish a pressurizer level ≥ 20%.	12 hours
	<u>AND</u>	
	AA.2.3 If in MODE 6, initiate action to be in MODE 6 with the water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately
BB. One channel inoperable.	BB.1 Place channel in bypass.	6 hours
	<u>AND</u>	
	BB.2 Continuously monitor hot leg level.	6 hours

SURVEILLANCE REQUIREMENTS

- NOTE -

Refer to Table 3.3.2-1 to determine which SRs apply for each Engineered Safety Features (ESF) Function.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.2.3	---------- **- NOTE -** Verification of setpoint not required for manual initiation functions. ---------- Perform TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT).	24 months
SR 3.3.2.4	---------- **- NOTE -** This surveillance shall include verification that the time constants are adjusted to the prescribed values. ---------- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.2.5	Perform CHANNEL OPERATIONAL TEST (COT).	92 days
SR 3.3.2.6	Verify ESFAS RESPONSE TIMES are within limit.	24 months on a STAGGERED TEST BASIS
SR 3.3.2.7	---------- **- NOTE -** This Surveillance is not required to be performed for actuated equipment which is included in the Inservice Test (IST) Program. ---------- Perform ACTUATION DEVICE TEST.	24 months
SR 3.3.2.8	Perform ACTUATION DEVICE TEST for squib valves.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.9	Perform ACTUATION DEVICE TEST for pressurizer heater circuit breakers.	24 months

Table 3.3.2-1 (page 1 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Safeguards Actuation						
a. Manual Initiation	1,2,3,4	2 switches	E,O	SR 3.3.2.3	NA	NA
	5	2 switches	G,Y	SR 3.3.2.3	NA	NA
b. Containment Pressure – High 2	1,2,3,4	4	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 6.21 psig	6.2 psig
c. Pressurizer Pressure – Low	1,2,3[(a)]	4	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1794.9 psig	1795.3 psig
d. Steam Line Pressure – Low	1,2,3[(a)]	4 per steam line	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 559.7 psig[(b)]	560.3 psig[(b)]
e. RCS Cold Leg Temperature (T_{cold}) – Low	1,2,3[(a)]	4 per loop	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 504.9°F ≤ 505.1°F	505°F

(a) Above the P-11 (Pressurizer Pressure) interlock, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F.

(b) Time constants used in the lead/lag controller are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds.

Table 3.3.2-1 (page 2 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
2. Core Makeup Tank (CMT) Actuation						
a. Manual Initiation	1,2,3,4$^{(j)}$	2 switches	E,N	SR 3.3.2.3	NA	NA
	4$^{(n)}$, 5$^{(l)}$	2 switches	E,U	SR 3.3.2.3	NA	NA
b. Pressurizer Water Level – Low 2	1,2,3,4$^{(j)}$	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 9.95% span ≤ 10.05% span	10% span
	4$^{(n)}$, 5$^{(l)}$	4	B,V	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 9.95% span ≤ 10.05% span	10% span
c. Safeguards Actuation	1,2,3,4,5$^{(l)}$	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.				
d. ADS Stages 1, 2, & 3 Actuation	1,2,3,4,5$^{(l)}$	Refer to Function 9 (ADS Stages 1, 2 & 3 Actuation) for all initiating functions and requirements.				
3. Containment Isolation						
a. Manual Initiation	1,2,3,4	2 switches	E,O	SR 3.3.2.3	NA	NA
	5$^{(m)}$, 6$^{(m)}$	2 switches	G,Y	SR 3.3.2.3	NA	NA
b. Manual Initiation of Passive Containment Cooling	1,2,3,4,5$^{(e,m)}$, 6$^{(e,m)}$	Refer to Function 12.a (Passive Containment Cooling Actuation) for initiating functions and requirements.				
c. Safeguards Actuation	1,2,3,4,5$^{(m)}$	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.				

(e) With decay heat > 9.0 MWt.

(l) With the RCS pressure boundary intact.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation Functions whose associated flow path is isolated.

(n) With the RCS being cooled by the RNS.

Table 3.3.2-1 (page 3 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation						
a. Manual Initiation	1,2(l),3(l),4(l)	2 switches	E,S	SR 3.3.2.3	NA	NA
b. Containment Pressure – High 2	1,2(l),3(l),4(l)	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 6.21 psig	6.2 psig
c. Steam Line Pressure						
(1) Steam Line Pressure – Low	1,2(l),3(a,l)	4 per steam line	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 559.7 psig(b)	560.3 psig(b)
(2) Steam Line Pressure – Negative Rate – High	3(d,l)	4 per steam line	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 100.1 psig with time constant ≥ 50 seconds	100 psig with time constant ≥ 50 seconds
d. T_{cold} – Low	1,2(l),3(a,l)	4 per loop	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 504.9°F ≤ 505.1°F	505°F
5. Turbine Trip						
a. Manual Main Feedwater Isolation	1,2	Refer to Function 6.a (Manual Main Feedwater Control Valve Isolation) for requirements.				
b. SG Narrow Range Water Level – High 2	1,2	4 per SG	B,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 82.05% span	82% span
c. Safeguards Actuation	1,2	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.				
d. Reactor Trip	1,2	Refer to Function 18.a (ESFAS Interlocks, Reactor Trip, P-4) for requirements.				

(a) Above the P-11 (Pressurizer Pressure) interlock, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F.

(b) Time constants used in the lead/lag controller are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds.

(d) Below the P-11 (Pressurizer Pressure) interlock.

(l) Not applicable if all MSIVs are closed.

Table 3.3.2-1 (page 4 of 13)
Engineered Safeguards Actuation System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6.	Main Feedwater Control Valve Isolation						
	a. Manual Initiation	1,2,3,4(m)	2 switches	E,S	SR 3.3.2.3	NA	NA
	b. SG Narrow Range Water Level – High 2	1,2,3,4(j,m)	4 per SG	B,R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 82.05% span	82% span
	c. Safeguards Actuation	1,2,3,4(m)	Refer to Function 1 (Safeguards Actuation) for all initiating functions and requirements.				
	d. Reactor Coolant Average Temperature (T_{avg}) – Low 1	1,2	4	B,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 549.9°F	550°F
	Coincident with Reactor Trip	1,2	Refer to Function 18.a (ESFAS Interlocks, Reactor Trip, P-4) for requirements.				
7.	Main Feedwater Pump Trip and Valve Isolation						
	a. Manual Initiation	Refer to Function 6.a (Manual Main Feedwater Control Valve Isolation) for requirements.					
	b. SG Narrow Range Water Level – High 2	1,2,3,4(j,m)	4 per SG	B,R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 82.05% span	82% span
	c. Safeguards Actuation	1,2,3,4(m)	Refer to Function 1 (Safeguards Actuation) for all initiating functions and requirements.				
	d. Reactor Coolant Average Temperature T_{avg} – Low 2	1,2	2 per loop	B,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 541.9°F	≥ 542°F
	Coincident with Reactor Trip	1,2	Refer to Function 18.a (ESFAS Interlocks, Reactor Trip, P-4) for requirements.				

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation Functions whose associated flow path is isolated.

Table 3.3.2-1 (page 5 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
8. Startup Feedwater Isolation						
a. SG Narrow Range Water Level – High 2	1,2,3,4[(o)]	4 per SG	B,S	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 82.05% span	82% span
b. T_{cold} – Low	1,2,3[(a)]	4 per loop	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 504.9°F ≤ 505.1°F	505°F
c. Manual Initiation	Refer to Function 6.a (Manual Main Feedwater Control Valve Isolation) for requirements.					
9. ADS Stages 1, 2 & 3 Actuation						
a. Manual Initiation	1,2,3,4	2 switch sets	E,O	SR 3.3.2.3	NA	NA
	5[(k)],6[(g,k)]	2 switch sets	G,X	SR 3.3.2.3	NA	NA
b. Core Makeup Tank (CMT) Level – Low 1	1,2,3,4	4 per tank	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 56.9% span	61.9% span
	5[(c,k)]	4 per OPERABLE tank	B,V	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 56.9% span	61.9% span
Coincident with CMT Actuation	Refer to Function 2 (CMT Actuation) for all initiating functions and requirements.					

(a) Above the P-11 (Pressurizer Pressure) interlock, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F.

(c) With pressurizer level ≥ 20%.

(g) With upper internals in place.

(o) Not applicable when the startup feedwater flow paths are isolated.

(k) Not applicable when the required ADS valves are open. See LCO 3.4.12 and LCO 3.4.13 for ADS valve and equivalent relief area requirements.

Table 3.3.2-1 (page 6 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
10. ADS Stage 4 Actuation						
a. Manual Initiation Coincident with	1,2,3,4	2 switch sets	E,O	SR 3.3.2.3	NA	NA
	$5^{(k)}$,$6^{(g,k)}$	2 switch sets	G,X	SR 3.3.2.3	NA	NA
RCS Wide Range Pressure – Low, or	1,2,3,4	4	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1198 psig	1200 psig
	$5^{(k)}$,$6^{(g,k)}$	4	B,X	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1198 psig	1200 psig
ADS Stages 1, 2 & 3 Actuation	Refer to Function 9 (Stages 1, 2, & 3 Actuation) for initiating functions and requirements					
b. CMT Level – Low 2	1,2,3,4	4 per tank	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 56.9% span	61.9% span
	$5^{(c,k)}$	4 per OPERABLE tank	B,V	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 56.9% span	61.9% span
Coincident with RCS Wide Range Pressure – Low, and	1,2,3,4	4	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1198 psig	1200 psig
	$5^{(c,k)}$	4	B,V	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1198 psig	1200 psig
Coincident with ADS Stages 1, 2 & 3 Actuation	1,2,3,4,$5^{(c,k)}$	Refer to Function 9 (ADS Stages 1, 2 & 3 Actuation) for initiating functions and requirements				
c. Coincident RCS Loop 1 and 2 Hot Leg Level – Low 2	$4^{(n)}$,$5^{(k)}$,$6^{(k)}$	1 per loop	BB,Y	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 2.98 in. above inside surface of the bottom of the hot legs	3 in. above inside surface of the bottom of the hot legs

(c) With pressurizer level ≥ 20%.

(g) With upper internals in place.

(k) Not applicable when the required ADS valves are open. See LCO 3.4.12 and LCO 3.4.13 for ADS valve and equivalent relief area requirements.

(n) With the RCS being cooled by the RNS.

Table 3.3.2-1 (page 7 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
11. Reactor Coolant Pump Trip						
a. ADS Stages 1, 2 & 3 Actuation	Refer to Function 9 (ADS Stages 1, 2 & 3 Actuation) for initiating functions and requirements.					
b. Reactor Coolant Pump Bearing Water Temperature – High	1,2	4 per RCP	B,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 190.4°F	190°F
c. Manual CMT Actuation	Refer to Function 2.a (Manual CMT Actuation) for requirements.					
d. Pressurizer Water Level – Low 2	1,2,3,4[(j)]	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 9.95% span ≤ 10.05% span	10% span
	4[(n)],5[(c,j)]	4	B,V	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 9.95% span ≤ 10.05% span	10% span
e. Safeguards Actuation	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.					
12. Passive Containment Cooling Actuation						
a. Manual Initiation	1,2,3,4	2 switches	E,O	SR 3.3.2.3	NA	NA
	5[(e)],6[(e)]	2 switches	G,Y	SR 3.3.2.3	NA	NA
b. Containment Pressure – High 2	1,2,3,4	4	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 6.21 psig	6.2 psig

(c) With pressurizer level ≥ 20%.

(e) With decay heat > 9.0 MWt.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(n) With the RCS being cooled by the RNS.

Table 3.3.2-1 (page 8 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
13. Passive Residual Heat Removal Heat Exchanger Actuation						
a. Manual Initiation	1,2,3,4	2 switches	E,O	SR 3.3.2.3	NA	NA
	5(l)	2 switches	E,U	SR 3.3.2.3	NA	NA
b. SG Narrow Range Water Level – Low	1,2,3,4(j)	4 per SG	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 20.95% span	21% span
Coincident with Startup Feedwater Flow – Low	1,2,3,4(j)	2 per feedwater line	H,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 198.8 gpm per SG	200 gpm per SG
c. SG Wide Range Water Level – Low	1,2,3,4(j)	4 per SG	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 53.95% span	54% span
d. ADS Stages 1, 2 & 3 Actuation	1,2,3,4,5(l)	Refer to Function 9 (ADS Stages 1, 2 & 3 Actuation) for initiating functions and requirements.				
e. CMT Actuation	Refer to Function 2 (CMT Actuation) for initiating functions and requirements.					
f. Pressurizer Water Level, High 3	1,2,3,4(j,p)	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 71.05% span	71% span

(l) With the RCS pressure boundary intact.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(p) Above the P-19 (RCS Pressure) interlock.

Table 3.3.2-1 (page 9 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
14. SG Blowdown Isolation						
a. Passive Residual Heat Removal Heat Exchanger Actuation	1,2,3,4[j,m]	Refer to Function 13 (Passive Residual Heat Removal Heat Exchanger Actuation) for all initiating functions and requirements.				
b. SG Narrow Range Water Level – Low	1,2,3,4[jj,m]	4 per SG	B,R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 20.95% span	21% span
15. Boron Dilution Block						
a. Source Range Neutron Flux Doubling	2[f],3[f],4[m]	4	B,T	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ Source Range Flux X 2.201 in 50 minutes	Source Range Flux X 2.2 in 50 minutes
	5[m]	4	B,P	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ Source Range Flux X 2.201 in 50 minutes	Source Range Flux X 2.2 in 50 minutes
b. Reactor Trip	Refer to Function 18.a (ESFAS Interlocks, Reactor Trip, P-4) for all requirements.					
16. Chemical Volume and Control System Makeup Isolation						
a. SG Narrow Range Water Level – High 2	1,2,3[m],4[j,m]	4 per SG	B,R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 82.05% span	82% span
b. Pressurizer Water Level – High 1	1,2,3[m]	4	B,Q	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 23.05% span	23% span
Coincident with Safeguards Actuation	1,2,3[m]	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.				
c. Pressurizer Water Level – High 2	1,2,3,4[j,m,p]	4	B,T	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤59.05% span	59% span
d. Containment Radioactivity – High 2	1,2,3[m]	4	B,Q	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 150 R/hr	100 R/hr
e. Manual Initiation	1,2,3[m],4[j,m]	2 switches	E,R	SR 3.3.2.3	NA	NA
f. Source Range Neutron Flux Doubling	Refer to Function 15.a (Boron Dilution Block, Source Range Neutron Flux Doubling) for all requirements.					

(f) Not applicable when critical or during intentional approach to criticality.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation Functions whose associated flow path is isolated.

(p) Above the P-19 (RCS Pressure) interlock.

Table 3.3.2-1 (page 10 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
17. Normal Residual Heat Removal System Isolation						
a. Containment Radioactivity – High 2	1,2,3(m)	4	B,Q	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 150 R/hr	100 R/hr
b. Safeguards Actuation	1,2,3(m)	Refer to Function 1 (Safeguards Actuation) for all initiating functions and requirements.				
c. Manual Initiation	1,2,3(m)	2 switch sets	E,Q	SR 3.3.2.3	NA	NA
18. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	3 divisions	D,M	SR 3.3.2.3	NA	NA
b. Pressurizer Pressure, P-11	1,2,3	4	J,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5	≤ 1970.4 psig	1970 psig
c. Intermediate Range Neutron Flux, P-6	2	4	J,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5	≥ 9.91 E-6% RTP	1E-5% RTP
d. Pressurizer Level, P-12	1,2,3,4,5,6	4	J,M BB,Y	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5	≤ 16.05% span	16% span
e. RCS Pressure, P-19	1,2,3,4(j)	4	J,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5	≤ 702 psig	700 psig
f. Reactor Trip Breaker Open, P-3	1,2,3	3 divisions	D,M	SR 3.3.2.3	NA	NA
19. Containment Air Filtration System Isolation						
a. Containment Radioactivity – High 1	1,2,3,4(j)	4	B,Z	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 3 R/hr	2 R/hr
b. Containment Isolation	Refer to Function 3 (Containment Isolation) for initiating functions and requirements.					

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation Functions whose associated flow path is isolated.

Table 3.3.2-1 (page 11 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
20. Main Control Room Isolation and Air Supply Initiation						
a. Control Room Air Supply Radiation – High 2	1,2,3,4	2	F,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 1.5×10^{-6} curies/m^3 DOSE EQUIVALENT I-131	1×10^{-6} curies/m^3 DOSE EQUIVALENT I-131
	Note (h)	2	G,K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 1.5×10^{-6} curies/m^3 DOSE EQUIVALENT I-131	1×10^{-6} curies/m^3 DOSE EQUIVALENT I-131
b. Pressurizer Pressure – Low	1,2,3[(a)]	4	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1794.9 psig	1795.3 psig
21 Auxiliary Spray and Purification Line Isolation						
a. Pressurizer Water Level – Low 1	1,2	4	B,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 19.95% span	20.0% span
b. Manual Initiation	1,2	Refer to Function 16.e (Manual Chemical Volume Control System (Makeup Isolation) for requirements.				
22. In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valve Actuation						
a. Manual Initiation	1,2,3,4[(j)]	2 switch sets	E,N	SR 3.3.2.3	NA	NA
	4[(n)],5,6	2 switch sets	G,Y	SR 3.3.2.3	NA	NA
b. ADS 4th Stage Actuation	Refer to Function 10 (ADS 4th Stage Actuation) for initiating functions and requirements.					

(a) Above the P-11 (Pressurizer Pressure) interlock, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F.

(h) During movement of irradiated fuel assemblies.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(n) With the RCS being cooled by the RNS.

Table 3.3.2-1 (page 12 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
23. IRWST Containment Recirculation Valve Actuation						
a. Manual Initiation	1,2,3,4$^{(j)}$	2 switch sets	E,N	SR 3.3.2.3	NA	NA
	4$^{(n)}$,5,6	2 switch sets	G,Y	SR 3.3.2.3	NA	NA
b. ADS Stage 4 Actuation	Refer to Function 10 (ADS Stage 4 Actuation) for all initiating functions and requirements.					
Coincident with IRWST Level – Low 3	1,2,3,4$^{(j)}$	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ Containment Elevation @ 109.99 ft.	Containment Elevation @ 110 ft.
	4$^{(n)}$,5$^{(k)}$,6$^{(k)}$	4	I,Y	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ Containment Elevation @ 109.99 ft.	Containment Elevation @ 110 ft.
24. Refueling Cavity Isolation						
a. Spent Fuel Pool Level – Low	6	3	H,P	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 39.74 ft.	39.75 ft.
25. ESF Coincidence Logic						
a. Coincidence Logic	1,2,3,4	4 divisions, 1 battery-backed subsystem per division	D,O	SR 3.3.2.2	NA	NA
	5,6	4 divisions, 1 battery-backed subsystem per division	G,W	SR 3.3.2.2	NA	NA

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(k) Not applicable when the required ADS valves are open. See LCO 3.4.12 and LCO 3.4.13 for ADS valve and equivalent relief area requirements.

(n) With the RCS being cooled by the RNS.

Table 3.3.2-1 (page 13 of 13)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
26. ESF Actuation						
a. ESF Actuation Subsystem	1,2,3,4	4 divisions, 1 battery-backed subsystem per division	D,O	SR 3.3.2.2 SR 3.3.2.7 SR 3.3.2.8	NA	NA
	5,6	4 divisions, 1 battery-backed subsystem per division	G,W	SR 3.3.2.2 SR 3.3.2.7	NA	NA
27. Pressurizer Heater Trip						
a. Core Makeup Tank Actuation	1,2,3,4$^{(j,p)}$	Refer to Function 2 (Core Makeup Tank Actuation) for all initiating functions and requirements. In addition to the requirements for Function 2, SR 3.3.2.9 also applies.				
b. Pressurizer Water Level, High 3	1,2,3,4$^{(j,p)}$	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 71.05%	71%
28. Chemical and Volume Control System Letdown Isolation						
a. Hot Leg Level – Low 1	4$^{(n,r)}$,5$^{(r)}$,6$^{(q,r)}$	1 per loop	C,AA	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 17.98 in. above inside surface of the bottom of the hot legs	18 in. above inside surface of the bottom of the hot legs
29. SG Power Operated Relief Valve and Block Valve Isolation						
a. Manual Initiation	1,2,3,4$^{(j)}$	2 switches	E,N	SR 3.3.2.3	NA	NA
b. Steam Line Pressure – Low	1,2,3,4$^{(j)}$	4 per steam line	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 559.7 psig	560.3 psig$^{(b)}$

(b) Time constants used in the lead/lag controller are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(n) With the RCS being cooled by the RNS.

(p) Above the P-19 (RCS Pressure) interlock.

(q) With the water level < 23 feet above the top of the reactor vessel flange.

(r) Below the P-12 (Pressurizer Level) interlock.

3.3 INSTRUMENTATION

3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

- NOTES -

1. LCO 3.0.4 not applicable.

2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel 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 5.6.7.	Immediately
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.3-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.3-1.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

- NOTE -

SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.3.2	----- **- NOTE -** Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	24 months

Table 3.3.3-1 (page 1 of 1)
Post-Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS/ DIVISIONS	CONDITION REFERENCED FROM REQUIRED ACTION D.1
1.	Neutron Flux (Intermediate Range)	2	E
2.	Reactor Coolant System (RCS) Hot Leg Temperature (Wide Range)	2	E
3.	RCS Cold Leg Temperature (Wide Range)	2	E
4.	RCS Pressure (Wide Range)	2	E
5.	Pressurizer Pressure and RCS Subcooling Monitor[(a)]	2	E
6.	Containment Water Level	2	E
7.	Containment Pressure	2	E
8.	Containment Pressure (Extended Range)	2	E
9.	Containment Area Radiation (High Range)	2	E
10.	Pressurizer Level and Associated Reference Leg Temperature	2	E
11.	IRWST Water Level	2	E
12.	PRHR Flow and PRHR Outlet Temperature	2 flow & 1 temperature	E
13.	Core Exit Temperature -- Quadrant 1	2[(b)]	E
14.	Core Exit Temperature -- Quadrant 2	2[(b)]	E
15.	Core Exit Temperature -- Quadrant 3	2[(b)]	E
16.	Core Exit Temperature -- Quadrant 4	2[(b)]	E
17.	PCS Storage Tank Level and PCS Flow	2 level & 1 flow	E
18.	Remotely Operated Containment Isolation Valve Position	1/valve[(c)]	E
19.	IRWST to RNS Suction Valve Status	2	E

(a) RCS Subcooling calculated from pressurizer pressure and RCS hot leg temperature.

(b) A channel consists of two thermocouples within a single division. Each quadrant contains two divisions. The minimum requirement is two OPERABLE thermocouples in each of the two divisions.

(c) 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.

3.3 INSTRUMENTATION

3.3.4 Remote Shutdown Workstation (RSW)

LCO 3.3.4 The Remote Shutdown Workstation (RSW) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and
MODE 4 with RCS average temperature (T_{avg}) ≥ 350°F.

ACTIONS

- NOTE -

LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RSW inoperable.	A.1	Restore to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4 with T_{avg} < 350°F.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Verify each required transfer switch is capable of performing the required function.	24 months
SR 3.3.4.2	Verify that the RSW communicates indication and controls with Division A, B, C and D of the PMS.	24 months
SR 3.3.4.3	Verify the OPERABILITY of the RSW hardware and software.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.4	Perform TADOT of the reactor trip breaker open/closed indication.	24 months

3.3 INSTRUMENTATION

3.3.5 Diverse Actuation System (DAS) Manual Controls

LCO 3.3.5 The DAS manual controls for each function in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more manual DAS controls inoperable.	A.1	Restore DAS manual controls to OPERABLE status.	30 days
B. Completion Time of Required Action A not met for inoperable DAS manual reactor trip control.	B.1	Perform SR 3.3.1.6.	Once per 31 days on a STAGGERED TEST BASIS
	<u>AND</u>		
	B.2	Restore all controls to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry
C. Completion Time of Required Action A not met for inoperable DAS manual actuation control other than reactor trip.	C.1	Perform SR 3.3.2.2.	Once per 31 days on a STAGGERED TEST BASIS
	<u>AND</u>		
	C.2	Restore all controls to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry
D. Completion Time of Required Action B not met. <u>OR</u> Completion Time of Required Action C 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 3.3.5.1	-------------------- **- NOTE -** Verification of setpoint not required. --------------------	
	Perform TRIP ACTUATION DEVICE OPERATIONAL TEST (TADOT).	24 months

Table 3.3.5-1 (page 1 of 1)
DAS Manual Controls

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CONTROLS
1.	Reactor trip manual controls	1,2	2 switches
2.	PRHR HX control and IRWST gutter control valves	1,2,3,4,5(a)	2 switches
3.	CMT isolation valves	1,2,3,4,5(a)	2 switches
4.	ADS stage 1 valves	1,2,3,4,5(a)	2 switches
5.	ADS stage 2 valves	1,2,3,4,5(a)	2 switches
6.	ADS stage 3 valves	1,2,3,4,5(a)	2 switches
7.	ADS stage 4 valves	1,2,3,4,5,6(c)	2 switches
8.	IRWST injection squib valves	1,2,3,4,5,6	2 switches
9.	Containment recirculation valves	1,2,3,4,5,6	2 switches
10.	Passive containment cooling drain valves	1,2,3,4,5(b),6(b)	2 switches
11.	Selected containment isolation valves	1,2,3,4,5,6	2 switches

(a) With RCS pressure boundary intact.

(b) With the calculated reactor decay heat > 9.0 MWt.

(c) In MODE 6 with reactor internals in place.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer Pressure is greater than or equal to the limit specified in the COLR

b. RCS Average Temperature is less than or equal to the limit specified in the COLR, and

c. RCS total flow rate ≥ 301,670 gpm and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1.

- NOTE -

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 RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	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 pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 301,670 gpm and greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	--- **- NOTE -** Not required to be performed until 24 hours after ≥ 90% RTP. ---	
	Verify that RCS total flow rate is ≥ 301,670 gpm and greater than or equal to the limit 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 temperature (T_{avg}) shall be ≥ 551°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with k_{eff} < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} in each loop ≥ 551°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 parameters to within limits. AND 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. AND B.2 Be in MODE 4 with RCS pressure < 500 psig.	6 hours 24 hours
C. -------------------------------- **- NOTE -** Required Action C.2 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. AND C.2 Determine RCS is acceptable for continued operation.	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 inservice leak and hydrostatic testing. ---	
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates within limits specified in the PTLR.	30 minutes

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops

LCO 3.4.4 Two RCS loops shall be OPERABLE and in operation (Four Reactor Coolant Pumps (RCPs) operating with variable speed control bypassed).

- NOTES -

1. No RCP shall be started when the reactor trip breakers are closed.

2. No RCP shall be started when the RCS temperature is ≥ 200°F unless pressurizer level is < 92%.

3. No RCP shall be started with any RCS cold leg temperature ≤ 200°F unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures.

4. All RCPs may be de-energized in MODE 3, 4, or 5 for ≤ 1 hour per 8 hour period provided:

    a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

    b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5, whenever the reactor trip breakers are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------------------------- **- NOTE -** Required Action A.1 must be completed whenever Condition A is entered. -------------------------------- Requirements of LCO not met in MODE 1 or 2.	A.1 Be in MODE 3 with the reactor trip breakers open.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------------------------------- **- NOTE -** Required Action B.1 must be completed whenever Condition B is entered. -------------------------------- Requirements of LCO not met in MODE 3, 4, or 5.	B.1	Be in MODE 3, 4, or 5 with the reactor trip breakers open.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each RCS loop is in operation with variable speed control bypassed.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 Pressurizer

LCO 3.4.5 The pressurizer water level shall be ≤ 92% of span.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Restore pressurizer water level within limit.	6 hours
	<u>OR</u>	
	A.2.1 Be in MODE 3 with reactor trip breakers open.	6 hours
	<u>AND</u>	
	A.2.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify pressurizer water level ≤ 92% of span.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 Pressurizer Safety Valves

LCO 3.4.6 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RNS isolated or RCS temperature ≥ 275°F.

- NOTE -

The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions.

This exception is allowed for 36 hours following entry into MODE 3, provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. <u>OR</u> Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with RNS aligned to the RCS and RCS temperature < 275°F.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify each pressurizer safety valve 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)

3.4.7 RCS Operational LEAKAGE

LCO 3.4.7 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 0.5 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE from the RCS,

d. 150 gallons per day primary to secondary LEAKAGE through any one SG, and

e. 500 gallons per day primary to IRWST LEAKAGE through the passive residual heat removal heat exchanger (PRHR HX).

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 not met. <u>OR</u> Pressure boundary LEAKAGE exists. OR 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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	-- **- 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 3.4.7.2	-- **- 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 any one SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Minimum RCS Flow

LCO 3.4.8 At least one Reactor Coolant Pump (RCP) shall be in operation with a total flow through the core of at least 3,000 gpm.

- NOTES -

1. All RCPs may be de-energized for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. No RCP shall be started when the RCS temperature is ≥ 200°F unless pressurizer level is < 92%.

3. No RCP shall be started with any RCS cold leg temperature ≤ 200°F unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures.

APPLICABILITY: MODES 3, 4, and 5, whenever the reactor trip breakers are open and with unborated water sources not isolated from the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No RCP in operation.	A.1 Isolate all sources of unborated water.	1 hour
	AND	
	A.2 Perform SR 3.1.1.1, (SDM verification).	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify that at least one RCP is in operation at ≥ 10% rated speed or equivalent.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Leakage Detection Instrumentation

LCO 3.4.9 The following RCS leakage detection instrumentation shall be OPERABLE:

a. Two containment sump level channels;

b. One containment atmosphere radioactivity monitor (gaseous N13/F18).

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

- NOTES -

1. The N13/F18 containment atmosphere radioactivity monitor is only required to be OPERABLE in MODE 1 with RTP > 20%.

2. Containment sump level measurements cannot be used for leak detection if leakage is prevented from draining to the sump such as by redirection to the IRWST by the containment shell gutter drains.

ACTIONS

- NOTE

LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required containment sump channel inoperable.	A.1	Verify that the volume input per day to the containment sump does not change (+ or -) more than 10 gallons or 33% of the volume input (whichever is greater). The volume used for comparison will be the value taken during the first day following the entrance into this CONDITION.	Once per 24 hours
	AND		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Restore two containment sump channels to OPERABLE status.	14 days
B. Two required containment sump channels inoperable.	B.1	---------------------------------- **- NOTE -** Not required until 12 hours after establishment of steady state operation. ----------------------------------	
		Perform SR 3.4.7.1 (RCS inventory balance).	Once per 24 hours
	<u>AND</u>		
	B.2	Restore one containment sump channel to OPERABLE status.	72 hours
C. Required containment atmosphere radioactivity monitor inoperable.	C.1.1	Analyze grab samples of containment atmosphere.	Once per 24 hours
	<u>OR</u>		
	C.1.2	---------------------------------- **- NOTE -** Not required until 12 hours after establishment of steady state operation. ----------------------------------	
		Perform SR 3.4.7.1.	Once per 24 hours
	<u>AND</u>		
	C.2	Restore containment atmosphere radioactivity monitor to OPERABLE status.	30 days
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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. All required monitors inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Perform a CHANNEL CHECK of required containment atmosphere radioactivity monitor.	12 hours
SR 3.4.9.2	Perform a COT of required containment atmosphere radioactivity monitor.	92 days
SR 3.4.9.3	Perform a CHANNEL CALIBRATION of required containment sump monitor.	24 months
SR 3.4.9.4	Perform a CHANNEL CALIBRATION of required containment atmosphere radioactivity monitor.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 RCS Specific Activity

LCO 3.4.10 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	A.1	------------------------------------ **- NOTE -** LCO 3.0.4 is not applicable. ------------------------------------ Verify DOSE EQUIVALENT I-131 to be ≤ 60 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 > 280 μCi/gm.	B.1	Perform SR 3.4.10.2.	4 hours
	<u>AND</u>		
	B.2	Be in MODE 3 with T_{avg} < 500°F.	6 hours
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 > 60 μCi/gm.	C.1	Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 280 μCi/gm.	7 days
SR 3.4.10.2	-- **- NOTE -** Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days <u>AND</u> Between 2 to 6 hours after a THERMAL POWER change of ≥ 15% of RTP within a 1 hour period

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Automatic Depressurization System (ADS) – Operating

LCO 3.4.11 The ADS, including 10 flow paths, shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One flow path inoperable.	A.1 Restore flow path(s) to OPERABLE status.	72 hours
B. One stage 1 ADS flow path inoperable. AND Either one stage 2 ADS flow path inoperable or one stage 3 ADS flow path inoperable.	B.1 Restore flow path(s) to OPERABLE status.	72 hours
C. Required Action and associated Completion Time not met. OR Requirements of LCO not met for reasons other than Condition A or B.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify that the motor operated valve in series with each 4th stage ADS valve is fully open.	12 hours
SR 3.4.11.2	Verify that each stage 1, 2, and 3 ADS valve is OPERABLE by stroking them open.	In accordance with the Inservice Testing Program
SR 3.4.11.3	Verify that each stage 4 ADS valve is OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Automatic Depressurization System (ADS) – Shutdown, RCS Intact

LCO 3.4.12 The ADS, including 9 flow paths, shall be OPERABLE.

APPLICABILITY: MODE 5 with RCS pressure boundary intact.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required flow path inoperable.	A.1 Restore flow path(s) to OPERABLE status.	72 hours
B. One required stage 1 ADS flow path inoperable. <u>AND</u> Either one required stage 2 or stage 3 ADS flow path inoperable.	B.1 Restore flow path(s) to OPERABLE status.	72 hours
C. Required Action and associated Completion Time not met. <u>OR</u> Requirements of LCO not met for reasons other than Condition A.	C.1 Initiate action to be in MODE 5, with RCS open and ≥ 20% pressurizer level.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	For flow paths required to be OPERABLE, the SRs of LCO 3.4.11, “Automatic Depressurization System (ADS) – Operating” are applicable.	In accordance with applicable SRs

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 Automatic Depressurization System (ADS) – Shutdown, RCS Open

LCO 3.4.13 ADS stage 1, 2, and 3, flow paths shall be open.
ADS stage 4 with 2 flow paths shall be OPERABLE.

- NOTE -

In MODE 5, the ADS valves may be closed to facilitate RCS vacuum fill operations to establish a pressurizer level ≥ 20%, provided ADS valve OPERABILITY meets LCO 3.4.12, ADS – Shutdown, RCS Intact.

APPLICABILITY: MODE 5 with RCS pressure boundary open or pressurizer level < 20%;
MODE 6 with upper internals in place.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ADS stage 1, 2, or 3 flow path closed.	A.1 Open the affected flow path.	72 hours
	OR	
	A.2 Open an alternative flow path with an equivalent area.	72 hours
B. One required ADS stage 4 flow path closed and inoperable.	B.1 Open an alternative flow path with an equivalent area.	36 hours
	OR	
	B.2 Restore two ADS stage 4 flow paths to OPERABLE status.	36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time not met while in MODE 5. OR Requirements of LCO not met for reasons other than Conditions A or B while in MODE 5.	C.1	Initiate action to fill the RCS to establish ≥ 20% pressurizer level.	Immediately
	AND		
	C.2	Suspend positive reactivity additions.	Immediately
D. Required Action and associated Completion Time not met while in MODE 6. OR Requirements of LCO not met for reasons other than Conditions A or B while in MODE 6.	D.1	Initiate action to remove the upper internals.	Immediately
	AND		
	D.2	Suspend positive reactivity additions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	Verify that each ADS stage 1, 2, and 3 valve is in the fully open position.	12 hours
SR 3.4.13.2	For each ADS stage 4 flow path required to be OPERABLE, the following SRs of LCO 3.4.11, “Automatic Depressurization System (ADS) – Operating” are applicable: SR 3.4.11.1 SR 3.4.11.3	In accordance with applicable SRs

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.14 At least one of the following Overpressure Protection Systems shall be OPERABLE, with the accumulators isolated:

a. The Normal Residual Heat Removal System (RNS) suction relief valve, or

b. The RCS depressurized and an RCS vent of ≥ 4.15 square inches.

- NOTE -

When the RCS temperature is ≥ 200°F, a reactor coolant pump (RCP) may not be started if the pressurizer level is ≥ 92%.

APPLICABILITY: MODE 4 when any cold leg temperature is ≤ 275°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

- NOTE -

Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. An accumulator not isolated when the accumulator pressure is ≥ to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	A.1 Isolate affected accumulator.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Increase RCS cold leg temperature to a level acceptable for the existing accumulator pressure allowed in the PTLR.	12 hours
	<u>OR</u>	
	B.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours
C. The RNS suction relief valve inoperable.	C.1 Restore the RNS suction relief valve to OPERABLE status.	12 hours
	<u>OR</u>	
	C.2 Depressurize RCS and establish RCS vent of ≥ 4.15 square inches.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	Verify each accumulator is isolated.	12 hours
SR 3.4.14.2	Verify both RNS suction isolation valves in one RNS suction flow path are open.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.14.3	-- **- NOTE -** Only required to be performed when complying with LCO 3.4.14.b. -- Verify RCS vent ≥ 4.15 square inches is open.	12 hours for unlocked-open vent <u>AND</u> 31 days for locked-open vent
SR 3.4.14.4	Verify the lift setting of the RNS suction relief valve.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Pressure Isolation Valve (PIV) Integrity

LCO 3.4.15 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, with the RCS not being cooled by the RNS.

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. Leakage from one or more RCS PIVs not within limit.	A.1 ------------------------------------ **- NOTE -** Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.15.1 and be in the reactor coolant pressure boundary or the high pressure portion of the system. ------------------------------------	
	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.	8 hours
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Verify a second OPERABLE PIV can meet the leakage limits. This valve is required to be a check valve, or a closed valve, if it isolates a line that penetrates containment.	72 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 3.4.15.1	Verify leakage of each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 and ≤ 2255 psig.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Reactor Vessel Head Vent (RVHV)

LCO 3.4.16 The Reactor Vessel Head Vent shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with the RCS not being cooled by the RNS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One flow path inoperable.	A.1 Restore flow path to OPERABLE status.	72 hours
B. Two flow paths inoperable.	B.1 Restore at least one flow path to OPERABLE status.	6 hours
C. Required Action and associated Completion Time not met. <u>OR</u> Requirements of LCO not met for reasons other than Conditions A or B.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4, with the RCS cooling provided by the RNS.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify that each RVHV valve is OPERABLE by stroking it open.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Chemical and Volume Control System (CVS) Makeup Isolation Valves

LCO 3.4.17 Two CVS Makeup Isolation Valves shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CVS makeup isolation valve inoperable.	A.1	Restore two CVS makeup isolation valves to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met. <u>OR</u> Two CVS makeup isolation valves inoperable.	B.1	------------------------------------ **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. ------------------------------------ Isolate the flow path from the CVS makeup pumps to the Reactor Coolant System by use of at least one closed manual or one closed and de-activated automatic valve.	 1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify two CVS makeup isolation valves are OPERABLE by stroking the valves closed.	In accordance with the Inservice Testing Program
SR 3.4.17.2	Verify closure time of each CVS makeup isolation valve is ≤ 30 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Steam Generator (SG) Tube Integrity

LCO 3.4.18 SG tube integrity shall be maintained.

AND

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 repair 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.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	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.	B.1 Be in MODE 3.	6 hours
OR	AND	
SG tube integrity not maintained.	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.18.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.1 Accumulators

LCO 3.5.1 Both accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 with RCS pressure > 1000 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration outside limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	8 hours if Condition C or E of LCO 3.5.2 has not been entered OR 1 hour if Condition C or E of LCO 3.5.2 has been entered
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 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 the borated water volume in each accumulator is ≥ 1667 cu. ft., and ≤ 1732 cu. ft.	12 hours
SR 3.5.1.3	Verify the nitrogen cover gas pressure in each accumulator is ≥ 637 psig and ≤ 769 psig.	12 hours
SR 3.5.1.4	Verify the boron concentration in each accumulator is ≥ 2600 ppm and ≤ 2900 ppm.	31 days <u>AND</u> ------------------------- **- NOTE -** Only required for affected accumulators. ------------------------- Once within 6 hours after each solution volume increase of ≥ 51 cu. ft., 3.0% that is not the result of addition from the in-containment refueling water storage tank
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when pressurizer pressure is ≥ 2000 psig.	31 days
SR 3.5.1.6	Verify system flow performance of each accumulator in accordance with the System Level OPERABILITY Testing Program.	10 years

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.2 Core Makeup Tanks (CMTs) – Operating

LCO 3.5.2 Both CMTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4 with the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CMT inoperable due to one CMT outlet isolation valve inoperable.	A.1 Restore outlet isolation valve to OPERABLE status.	72 hours
B. One CMT inoperable due to one or more parameters (water temperature, boron concentration) not within limits.	B.1 Restore water temperature or boron concentration to within limits.	72 hours
C. Two CMTs inoperable due to water temperature or boron concentration not within limits.	C.1 Restore water temperature or boron concentration to within limits for one CMT.	8 hours if Condition B of LCO 3.5.1 has not been entered <u>OR</u> 1 hour if Condition B of LCO 3.5.1 has been entered
D. One CMT inoperable due to presence of non-condensible gases in one high point vent.	D.1 Vent noncondensible gases.	24 hours
E. One CMT inoperable for reasons other than Condition A, B, C, or D.	E.1 Restore CMT to OPERABLE status.	8 hours if Condition B of LCO 3.5.1 has not been entered <u>OR</u> 1 hour if Condition B of LCO 3.5.1 has been entered

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time not met. OR LCO not met for reasons other than A, B, C, D, or E.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the temperature of the borated water in each CMT is < 120°F.	24 hours
SR 3.5.2.2	Verify the borated water volume in each CMT is ≥ 2500 cu. ft.	7 days
SR 3.5.2.3	Verify each CMT inlet isolation valve is fully open.	12 hours
SR 3.5.2.4	Verify the volume of noncondensible gases in each CMT inlet line is ≤ 0.2 ft^3.	24 hours
SR 3.5.2.5	Verify the boron concentration in each CMT is ≥ 3400 ppm, and ≤ 3700 ppm.	7 days
SR 3.5.2.6	Verify each CMT outlet isolation valve is OPERABLE by stroking it open.	In accordance with the Inservice Testing Program
SR 3.5.2.7	Verify system flow performance of each CMT in accordance with the System Level OPERABILITY Testing Program.	10 years

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.3 Core Makeup Tanks (CMTs) – Shutdown, RCS Intact

LCO 3.5.3 One CMT shall be OPERABLE.

APPLICABILITY: MODE 4 with the RCS cooling provided by the Normal Residual Heat Removal System (RNS),
MODE 5 with the RCS pressure boundary intact.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required CMT inoperable due to one outlet isolation valve inoperable.	A.1 Restore required isolation valve to OPERABLE status.	72 hours
B. Required CMT inoperable due to one or more parameters (water temperature, boron concentration) not within limits.	B.1 Restore water temperature or boron concentration to within limits.	72 hours
C. Required CMT inoperable for reasons other than A or B.	C.1 Restore required CMT to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met. <u>OR</u> LCO not met for reasons other than A, B, or C.	D.1 Initiate action to be in MODE 5 with RCS pressure boundary open and ≥ 20% pressurizer level.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	For the CMT required to be OPERABLE, the SRs of Specification 3.5.2, “Core Makeup Tanks (CMTs) – Operating” are applicable.	In accordance with applicable SRs

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.4 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating

LCO 3.5.4 The PRHR HX shall be OPERABLE.

- NOTE -

When any reactor coolant pumps (RCPs) are operating, at least one RCP must be operating in the loop with the PRHR HX, Loop 1.

APPLICABILITY: MODES 1, 2, 3, and 4 with the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One air operated outlet isolation valve inoperable.	A.1 Restore air operated outlet isolation valve to OPERABLE status.	72 hours
B. One air operated IRWST gutter isolation valve inoperable.	B.1 Restore air operated IRWST gutter isolation valve to OPERABLE status.	72 hours
C. Presence of non-condensible gases in the high point vent.	C.1 Vent noncondensible gases.	24 hours
D. Required Action and associated Completion Time of Conditions A, B, or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4 with the RCS cooling provided by the RNS.	6 hours 24 hours
E. LCO not met for reasons other than A, B, or C.	E.1 Restore PRHR HX to OPERABLE status.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. Required Action and associated Completion Time for Condition E not met.	F.1	------------------------------------ **- NOTE -** Prior to initiating actions to change to a lower MODE, verify that redundant means of providing SG feedwater are OPERABLE. If redundant means are not OPERABLE, suspend LCO 3.0.3 and all other LCO Required Actions requiring MODE changes until redundant means are restored to OPERABLE status. ------------------------------------ Be in MODE 3.	6 hours
	<u>AND</u>		
	F.2	------------------------------------ **- NOTE -** Prior to stopping the SG feedwater, verify that redundant means of cooling the RCS to cold shutdown conditions are OPERABLE. If redundant means are not OPERABLE, suspend LCO 3.0.3 and all other LCO Required Actions requiring MODE changes until redundant means are restored to OPERABLE status. ------------------------------------ Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify the outlet manual isolation valve is fully open.	12 hours
SR 3.5.4.2	Verify the inlet motor operated isolation valve is open.	12 hours
SR 3.5.4.3	Verify the volume of noncondensible gases in the PRHR HX inlet line is $\leq 0.9\ ft^3$.	24 hours
SR 3.5.4.4	Verify that power is removed from the inlet motor operated isolation valve.	31 days
SR 3.5.4.5	Verify both PRHR air operated outlet isolation valves and both IRWST gutter isolation valves are OPERABLE by stroking open the valves.	In accordance with the System Level Inservice Testing Program
SR 3.5.4.6	Verify PRHR HX heat transfer performance in accordance with the System Level OPERABILITY Testing Program.	10 years
SR 3.5.4.7	Verify by visual inspection that the IRWST gutters are not restricted by debris.	24 months

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.5 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Shutdown, RCS Intact

LCO 3.5.5 The PRHR HX shall be OPERABLE.

- NOTE -

When any reactor coolant pumps (RCPs) are operating, at least one RCP must be operating in the loop with the PRHR HX, Loop 1.

APPLICABILITY: MODE 4 with the RCS cooling provided by the Normal Residual Heat Removal System (RNS),
MODE 5 with the RCS pressure boundary intact and pressurizer level ≥ 20%.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One air operated outlet isolation valve inoperable.	A.1 Restore air operated outlet valve to OPERABLE status.	72 hours
B. One air operated IRWST gutter isolation valve inoperable.	B.1 Restore air operated IRWST gutter isolation valve to OPERABLE status.	72 hours
C. Presence of non-condensible gases in the high point vent.	C.1 Vent noncondensible gases.	24 hours
D. PRHR HX inoperable for reasons other than A, B, or C.	D.1 Restore PRHR HX to OPERABLE status.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met. <u>OR</u> LCO not met for reasons other than A, B, C, or D.	E.1 Initiate action to be in MODE 5 with the RCS pressure boundary open and > 20% pressurizer level.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	The SRs of Specification 3.5.4, “Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating” – are applicable.	In accordance with applicable SRs

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.6 In-containment Refueling Water Storage Tank (IRWST) – Operating

LCO 3.5.6 The IRWST, with two injection flow paths and two containment recirculation flow paths, shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One IRWST injection line actuation valve flow path inoperable. <u>OR</u> One containment recirculation line actuation valve flow path inoperable.	A.1 Restore the inoperable actuation valve flow path to OPERABLE status.	72 hours
B. IRWST boron concentration not within limits. <u>OR</u> IRWST borated water temperature not within limits. <u>OR</u> IRWST borated water volume < 100% and > 97% of limit.	B.1 Restore IRWST to OPERABLE status.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One motor operated IRWST isolation valve not fully open. OR Power is not removed from one or more motor operated IRWST isolation valves.	C.1 Restore motor operated IRWST isolation valve to fully open condition with power removed from both valves.	1 hour
D. Required Action and associated Completion Time not met. OR LCO not met for reasons other than A, B, or C.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.6.1	Verify the IRWST water temperature is < 120°F.	24 hours
SR 3.5.6.2	Verify the IRWST borated water volume is > 73,100 cu. ft.	24 hours
SR 3.5.6.3	Verify the IRWST boron concentration is ≥ 2600 ppm and ≤ 2900 ppm.	31 days AND Once within 6 hours after each solution volume increase of 15,000 gal

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.6.4	Verify each motor operated IRWST isolation valve is fully open.	12 hours
SR 3.5.6.5	Verify power is removed from each motor operated IRWST isolation valve.	31 days
SR 3.5.6.6	Verify each motor operated containment recirculation isolation valve is fully open.	31 days
SR 3.5.6.7	Verify each IRWST injection and containment recirculation squib valve is OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program
SR 3.5.6.8	Verify by visual inspection that the IRWST screens and the containment recirculation screens are not restricted by debris.	24 months
SR 3.5.6.9	Verify IRWST injection and recirculation system flow performance in accordance with the System Level OPERABILITY Testing Program.	10 years

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3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.7 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5

LCO 3.5.7 The IRWST, with one injection flow path and one containment recirculation flow path, shall be OPERABLE.

APPLICABILITY: MODE 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required motor operated containment recirculation isolation valve not fully open.	A.1 Open required motor operated containment recirculation isolation valve.	72 hours
B. IRWST boron concentration not within limits. OR IRWST borated water temperature not within limits. OR IRWST borated water volume < 100% and > 97% of limit.	B.1 Restore IRWST to OPERABLE status.	8 hours
C. Required motor operated IRWST isolation valve not fully open. OR Power is not removed from required motor operated IRWST isolation valve.	C.1 Restore required motor operated IRWST isolation valve to fully open condition with power removed.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time not met. OR LCO not met for reasons other than A, B, or C.	D.1	Initiate action to be in MODE 5 with the RCS pressure boundary intact and ≥ 20% pressurizer level.	Immediately
	AND		
	D.2	Suspend positive reactivity additions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.7.1	For the IRWST and flow paths required to be OPERABLE, the SRs of Specification 3.5.6, “In-containment Refueling Water Storage Tank (IRWST) – Operating” are applicable.	In accordance with applicable SRs

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.8 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6

LCO 3.5.8 The IRWST, with one injection flow path and one containment recirculation flow path, shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required motor operated containment recirculation isolation valve not fully open.	A.1 Open required motor operated containment recirculation isolation valve.	72 hours
B. IRWST and refueling cavity boron concentration not within limits. <u>OR</u> IRWST and refueling cavity borated water temperature not within limits. <u>OR</u> IRWST and refueling cavity borated water volume < 100% and > 97% of limit.	B.1 Restore IRWST to OPERABLE status.	8 hours
C. Required motor operated IRWST isolation valve not fully open. <u>OR</u> Power is not removed from required motor operated IRWST isolation valve.	C.1 Restore required motor operated IRWST isolation valve to fully open condition with power removed.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met. OR LCO not met for reasons other than A, B, or C.	D.1 Initiate action to be in MODE 6 with the water level ≥ 23 feet above the top of the reactor vessel flange. AND D.2 Suspend positive reactivity additions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.8.1	Verify the IRWST and refueling cavity water temperature is < 120°F.	24 hours
SR 3.5.8.2	Verify the IRWST and refueling cavity water total borated water volume is > 73,100 cu. ft.	24 hours
SR 3.5.8.3	Verify the IRWST and refueling cavity boron concentration is ≥ 2600 ppm and ≤ 2900 ppm.	31 days AND Once within 6 hours after each solution volume increase of 15,000 gal
SR 3.5.8.4	For the IRWST and flow paths required to be OPERABLE, the following SRs of Specification 3.5.6, “In-containment Refueling Water Storage Tank (IRWST) – Operating” are applicable: SR 3.5.6.4 SR 3.5.6.6 SR 3.5.6.8 SR 3.5.6.5 SR 3.5.6.7	In accordance with applicable SRs

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

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 acceptance criteria.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	A.1	------------------------------------ **- 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. ------------------------------------	
		Verify the OPERABLE door is closed in the affected air lock.	1 hour
	AND		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Lock the OPERABLE door closed in the affected air lock.	24 hours
	<u>AND</u>		
	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.	B.1	------------------------------------ **- 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. ------------------------------------	
		Verify an OPERABLE door is closed in the affected air lock.	1 hour
	<u>AND</u>		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours
	AND		
	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
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	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
	AND		
	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) 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.	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 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
B. -------------------------------- **- NOTE -** Only applicable to penetration flow paths with two containment isolation valves. -------------------------------- One or more penetration flow paths with two containment isolation valves inoperable.	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

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
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	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
	<u>AND</u>		
	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 administrative means. ----------------------------------- Verify that the affected penetration flow path is isolated.	Once per 31 days
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 3.6.3.1	Verify each 16 inch containment purge valve is closed, except when the 16 inch containment purge valves are open for pressure control, ALARA or air quality considerations for personnel containment entry, or for Surveillances which require the valves to be open.	31 days
SR 3.6.3.2	-- **- 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 3.6.3.3	-- **- 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 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 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	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 ≤ +1.0 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. AND 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 ≤ 120°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. <u>AND</u> 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 Passive Containment Cooling System (PCS) – Operating

LCO 3.6.6 The passive containment cooling system shall be OPERABLE, with all three water flow paths OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One passive containment cooling water flow path inoperable.	A.1 Restore flow path to OPERABLE status.	7 days
B. Two passive containment cooling water flow paths inoperable.	B.1 Restore flow paths to OPERABLE status.	72 hours
C. One or more water storage tank parameters (temperature and volume) not within limits.	C.1 Restore water storage tank to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met. <u>OR</u> LCO not met for reasons other than A, B, or C.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify the water storage tank temperature ≥ 40°F and ≤ 120°F.	7 days <u>AND</u> 24 hours when water storage tank temperature is verified ≤ 50°F or ≥ 100°F
SR 3.6.6.2	Verify the water storage tank volume ≥ 756,700 gallons.	7 days
SR 3.6.6.3	Verify each passive containment cooling system, power operated, and automatic valve in each flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.6.6.4	Verify each passive containment cooling system automatic valve in each 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 3.6.6.5	Verify the air flow path from the shield building annulus inlet to the exit is unobstructed and, that all air baffle sections are in place.	24 months
SR 3.6.6.6	Verify passive containment cooling system flow and water coverage performance in accordance with the System Level OPERABILITY Testing Program.	At first refueling <u>AND</u> 10 years

3.6 CONTAINMENT SYSTEMS

3.6.7 Passive Containment Cooling System (PCS) – Shutdown

LCO 3.6.7 The passive containment cooling system shall be OPERABLE with all three water flow paths OPERABLE.

APPLICABILITY: MODE 5 with the calculated reactor decay heat > 9.0 MWt,
MODE 6 with the calculated reactor decay heat > 9.0 MWt.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One passive containment cooling water flow path inoperable.	A.1 Restore flow path to OPERABLE status.	7 days
B. Two passive containment cooling water flow paths inoperable.	B.1 Restore flow paths to OPERABLE status.	72 hours
C. One or more water storage tank parameters (temperature and volume) not within limits.	C.1 Restore water storage tank to OPERABLE status.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time not met. OR LCO not met for reasons other than A, B, or C.	D.1.1	If in MODE 5, initiate action to be in MODE 5 with the RCS pressure boundary intact and ≥ 20% pressurizer level.	Immediately
	OR		
	D.1.2	If in MODE 6, initiate action to be in MODE 6 with the water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately
	AND		
	D.2	Suspend positive reactivity additions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	The SRs of Specification 3.6.6, "Passive Containment Cooling System – Operating" are applicable.	In accordance with applicable SRs

3.6 CONTAINMENT SYSTEMS

3.6.8 Containment Penetrations

LCO 3.6.8 The containment penetrations shall be in the following status:

a. The equipment hatches closed and held in place by four bolts or, if open, clear of obstructions such that the hatches can be closed prior to steaming into the containment.

b. One door in each air lock closed or, if open, the containment air locks shall be clear of obstructions such that they can be closed prior to steaming into the containment.

c. The containment spare penetrations, if open, shall be clear of obstructions such that the penetrations can be closed prior to steaming into the containment.

d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Isolation signal.

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Restore containment penetrations to required status.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met. OR LCO not met for reasons other than Condition A.	B.1.1	If in MODE 5, initiate action to be in MODE 5 with the RCS pressure boundary intact and ≥ 20% pressurizer level.	Immediately
	OR		
	B.1.2	If in MODE 6, initiate action to be in MODE 6 with the water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately
	AND		
	B.2	Suspend positive reactivity additions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Verify each required containment penetration is in the required status.	7 days
SR 3.6.8.2	--- **- NOTE -** Only required to be met for an open equipment hatch. --- Verify that the hardware, tools, equipment and power source necessary to install the equipment hatch are available.	Prior to hatch removal AND 7 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.8.3	--- **- NOTE -** Not required to be met for automatic isolation valve(s) in penetrations closed to comply with LCO 3.6.8.d.1. ---	
	Verify one automatic isolation valve in each open penetration providing direct access from the containment atmosphere to the outside atmosphere actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.9 pH Adjustment

LCO 3.6.9 The pH adjustment baskets shall contain ≥ 560 ft^3 of trisodium phosphate (TSP).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The volume of trisodium phosphate not within limit.	A.1 Restore volume of trisodium phosphate to within limit.	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 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.9.1	Verify that the pH adjustment baskets contain at least 560 ft^3 of TSP (Na_3PO_4-12 H_2O).	24 months
SR 3.6.9.2	Verify that a sample from the pH adjustment baskets provides adequate pH adjustment of the post-accident water.	24 months

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, 3,
MODE 4 with the RCS not being cooled by the RNS.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required MSSVs inoperable.	A.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	4 hours
	<u>AND</u>	
	A.2 ------------------------------------ **- NOTE -** Only required in MODE 1. ------------------------------------ Reduce the Power Range Neutron Flux – High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met. <u>OR</u> One or more steam generators with $\geq$ 5 MSSVs inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with the RCS cooling provided by the RNS.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-------------------- **- NOTE -** -------------------- Only required to be performed in MODES 1 and 2. -- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ±1%.	In accordance with the Inservice Testing Program

Table 3.7.1-1 (page 1 of 1)
OPERABLE MSSVs versus Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
5	60
4	46
3	32
2	18

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER		LIFT SETTING (psig ± 3%)
STEAM GENERATOR		
#1	#2	
V030A	V030B	1185
V031A	V031B	1196
V032A	V032B	1208
V033A	V033B	1219
V034A	V034B	1231
V035A	V035B	1242

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 The minimum combination of valves required for steam flow isolation shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2, 3, and 4 except when steam flow is isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A. Restore valve to OPERABLE status.	8 hours
B. One or more of the turbine stop valves and its associated turbine control valve, turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 1.	B. Restore valve to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two MSIVs inoperable in MODE 1. OR One MSIV inoperable and one or more of the turbine stop valves and its associated turbine control valve, all turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 1. OR Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 2.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. -------------------------------- **- NOTE -** Separate Condition entry is allowed for each MSIV. -------------------------------- One or two MSIVs inoperable in MODE 2, 3, or 4. <u>OR</u> One or more of the turbine stop valves and its associated turbine control valve, all turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 2, 3, or 4.	D.1 Isolate associated steam flow path. <u>AND</u> D.2 Verify flow path remains closed.	8 hours Once per 7 days
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 with the RCS cooling provided by the RNS.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-- **- NOTE -** Only required to be performed prior to entry into MODE 2. -- Verify MSIV closure time ≤ 5 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.2.2	-- **- NOTE -** Only required to be performed prior to entry into MODE 2. -- Verify turbine stop, turbine control, turbine bypass, and moisture separator reheater 2^{nd} stage steam isolation valves' closure time ≤ 5 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation and Control Valves (MFIVs and MFCVs)

LCO 3.7.3 The MFIV and the MFCV for each Steam Generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4 except when the MFIVs or associated MFCV are closed and deactivated.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two MFIVs inoperable.	A.1 Close or isolate the MFIV flow path. <u>AND</u> A.2 Verify MFIV is closed or isolated.	72 hours Once per 7 days
B. One or two MFCVs inoperable.	B.1 Close or isolate the MFCV the flow path. <u>AND</u> B.2 Verify MFCV is closed or isolated.	72 hours Once per 7 days
C. Two valves in the same flow path inoperable.	C.1 Isolate affected flow path.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4 with the RCS cooling provided by the RNS.	24 hours
	AND		
	D.3.1	Isolate the affected flow path(s).	36 hours
	OR		
	D.3.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	-- **- NOTE -** Only required to be performed prior to entry into MODE 2. --	
	Verify the closure time of each MFIV and MFCV is ≤ 5 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be ≤ 0.1 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 3.7.4.1	Verify the specific activity of the secondary coolant ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131.	31 days

3.7 PLANT SYSTEMS

3.7.5 Spent Fuel Pool Water Level

LCO 3.7.5 The spent fuel pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: At all times.

ACTIONS

- NOTE -

LCOs 3.0.3 and 3.0.8 are not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level < 23 ft.	A.1 Suspend movement of irradiated fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore water level to ≥ 23 ft.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify the spent fuel 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

3.7.6 Main Control Room Habitability System (VES)

LCO 3.7.6 The Main Control Room (MCR) Habitability System shall be OPERABLE.

- NOTE -

The MCR boundary may be opened intermittently under administrative control.

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

ACTIONS

- NOTE -

LCO 3.0.8 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One VES valve or damper inoperable.	A.1 Restore VES valve or damper to OPERABLE status.	7 days
B. MCR air temperature not within limit.	B.1 Restore MCR air temperature to within limit.	24 hours
C. Loss of integrity of MCR pressure boundary.	C.1 Restore MCR pressure boundary to OPERABLE status.	24 hours
D. Required Action and associated Completion Time of Conditions A, B, or C not met in MODE 1, 2, 3, or 4.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Required Action and associated Completion Time of Conditions A, B, or C not met during movement of irradiated fuel.	E.1	Suspend movement of irradiated fuel assemblies.	Immediately
F. VES inoperable in MODE 1, 2, 3, or 4.	F.1	Be in MODE 3.	6 hours
	AND		
	F.2	Be in MODE 4.	12 hours
	AND		
	F.3	Restore VES to OPERABLE status.	36 hours
G. VES inoperable during movement of irradiated fuel.	G.1	Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify Main Control Room air temperature is ≤ 75°F.	24 hours
SR 3.7.6.2	Verify that the compressed air storage tanks are pressurized to ≥ 3400 psig.	24 hours
SR 3.7.6.3	Verify that each VES air delivery isolation valve is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.4	Verify that each VES air header manual isolation valve is in an open position.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.6.5	Verify that the air quality of the air storage tanks meets the requirements of Appendix C, Table C-1 of ASHRAE Standard 62.	92 days
SR 3.7.6.6	Verify that all VBS Main Control Room isolation valves are OPERABLE and will close upon receipt of an actual or simulated actuation signal.	24 months
SR 3.7.6.7	Verify that each VES pressure relief isolation valve within the MCR pressure boundary is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.8	Verify that each VES pressure relief damper is OPERABLE.	24 months
SR 3.7.6.9	Verify that the self-contained pressure regulating valve in each VES air delivery flow path is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.10	Verify that one VES air delivery flow path maintains a 1/8-inch-water gauge positive pressure in the MCR envelope relative to the adjacent areas at the required air addition flow rate of 65 ± 5 scfm using the safety related compressed air emergency air storage tanks.	24 months

3.7 PLANT SYSTEMS

3.7.7 Startup Feedwater Isolation and Control Valves

LCO 3.7.7 Both Startup Feedwater Isolation Valves and Control Valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4 except when the startup feedwater flow paths are isolated.

ACTIONS

- NOTES -

1. Flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with one inoperable valve.	A.1 Isolate the affected flow path(s).	72 hours
	AND	
	A.2 Verify affected flow path(s) is isolated.	Once per 7 days
B. One flow path with two inoperable valves.	B.1 Isolate the affected flow path.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 4 with the RCS cooling provided by the RNS.	24 hours
	AND	
	C.3 Isolate the affected flow path(s).	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	Verify both startup feedwater isolation and control valves are OPERABLE.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.8 Main Steam Line Leakage

LCO 3.7.8 Main Steam Line leakage through the pipe walls inside containment shall be limited to 0.5 gpm.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Main Steam Line leakage exceeds operational limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify main steam line leakage into the containment sump ≤ 0.5 gpm.	Per SR 3.4.7.1

3.7 PLANT SYSTEMS

3.7.9 Fuel Storage Pool Makeup Water Sources

LCO 3.7.9 Fuel storage pool makeup water source shall be OPERABLE.

- NOTES -

1. OPERABILITY of the cask washdown pit is required when the calculated fuel storage pool decay heat ≥ 4.6 MWt and ≤ 5.4 MWt, and with the calculated reactor decay heat > 9 MWt.

2. OPERABILITY of the passive containment cooling water source is required when the calculated fuel storage pool decay heat > 5.4 MWt.

APPLICABILITY: During storage of fuel in the fuel storage pool with a calculated decay heat ≥ 4.6 MWt.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required fuel storage pool makeup water source inoperable.	A.1 ------------------------------------ **- NOTE -** LCOs 3.0.3 and 3.0.8 are not applicable. ------------------------------------ Initiate action to restore the required makeup water source to OPERABLE status.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify the passive containment cooling system water storage tank volume is ≥ 400,000 gallons.	7 days
SR 3.7.9.2	Verify the water level in the cask washdown pit is ≥ 13.75 ft.	30 days
SR 3.7.9.3	Verify the spent fuel storage pool makeup isolation valves PCS-PL-V009, PCS-PL-V045, PCS-PL-V051, SFS-PL-V042, SFS-PL-V045, SFS-PL-V049, SFS-PL-V066, and SFS-PL-V068 are OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.10 Steam Generator Isolation Valves

LCO 3.7.10 The steam generator isolation valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with the RCS not being cooled by the RNS.

ACTIONS

- NOTES -

1. Steam generator blowdown flow path(s) may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORV flow paths with one SG isolation valve inoperable.	A.1 Isolate the flow path by use of at least one closed and deactivated automatic valve.	72 hours
B. One or more blowdown flow paths with one SG isolation valve inoperable.	B.1 Isolate the flow path by one closed valve.	72 hours
	AND	
	B.2 Verify that the affected SG blowdown flow path is isolated.	Once per 7 days
C. One or more PORV flow paths with two SG isolation valves inoperable.	C.1 Isolate the affected flow path by use of at least one closed and deactivated automatic valve.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more blowdown flow paths with two SG isolation valves inoperable.	D.1	Isolate the flow path by one closed valve.	8 hours
	<u>AND</u>		
	D.2	Verify that the affected SG blowdown flow path is isolated.	Once per 7 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 4 with the RCS cooling provided by the RNS.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify each steam generator isolation valve (PORV block valves (SGS-PL-V027A & B), PORVs (SGS-PL-V233A & B), and blowdown isolation valves (SGS-PL-V074A & B and SGS-PL-V075A & B)) is OPERABLE by stroking the valve closed.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.11 Fuel Storage Pool Boron Concentration

LCO 3.7.11 The fuel storage pool boron concentration shall be ≥ 2300 ppm.

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

ACTIONS

- NOTE -

LCOs 3.0.3 and 3.0.8 are not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND	
	A.2.1 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately
	OR	
	A.2.2 Initiate action to perform a fuel storage pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify the fuel storage pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool Storage

LCO 3.7.12 The combination of initial enrichment, burnup, and decay time of each fuel assembly stored in Region 2 in an “All Cell” storage configuration shall be within the limits specified in Figure 3.7.12-1 or

The combination of initial enrichment, burnup, and decay time of each fuel assembly stored in Region 2 in the Spent Fuel (1.361 w/o) locations in a “1-out-of-4 5.0 weight percent fresh” configuration shall be within the limits specified in Figures 3.7.12-2.

Fuel may be stored in Region 2 in both “All Cell” and “1-out-of-4 5.0 weight-percent fresh” storage configurations together, provided the fuel stored in the interface locations surrounding the “1-out-of-4 5.0 weight-percent fresh” group(s) meet the initial enrichment, burnup, and decay time limits specified in Figure 3.7.12-2.

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

ACTIONS

- NOTE -

LCOs 3.0.3 and 3.0.8 are not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify by administrative means the initial enrichment, burnup, and decay time of the fuel assembly is in accordance with Figures 3.7.12-1 or 3.7.12-2, as applicable for “All Cell,” “1-out-of-4 5.0 weight-percent fresh” and interface spent fuel assembly storage configurations.	Prior to storing the fuel assembly in Region 2

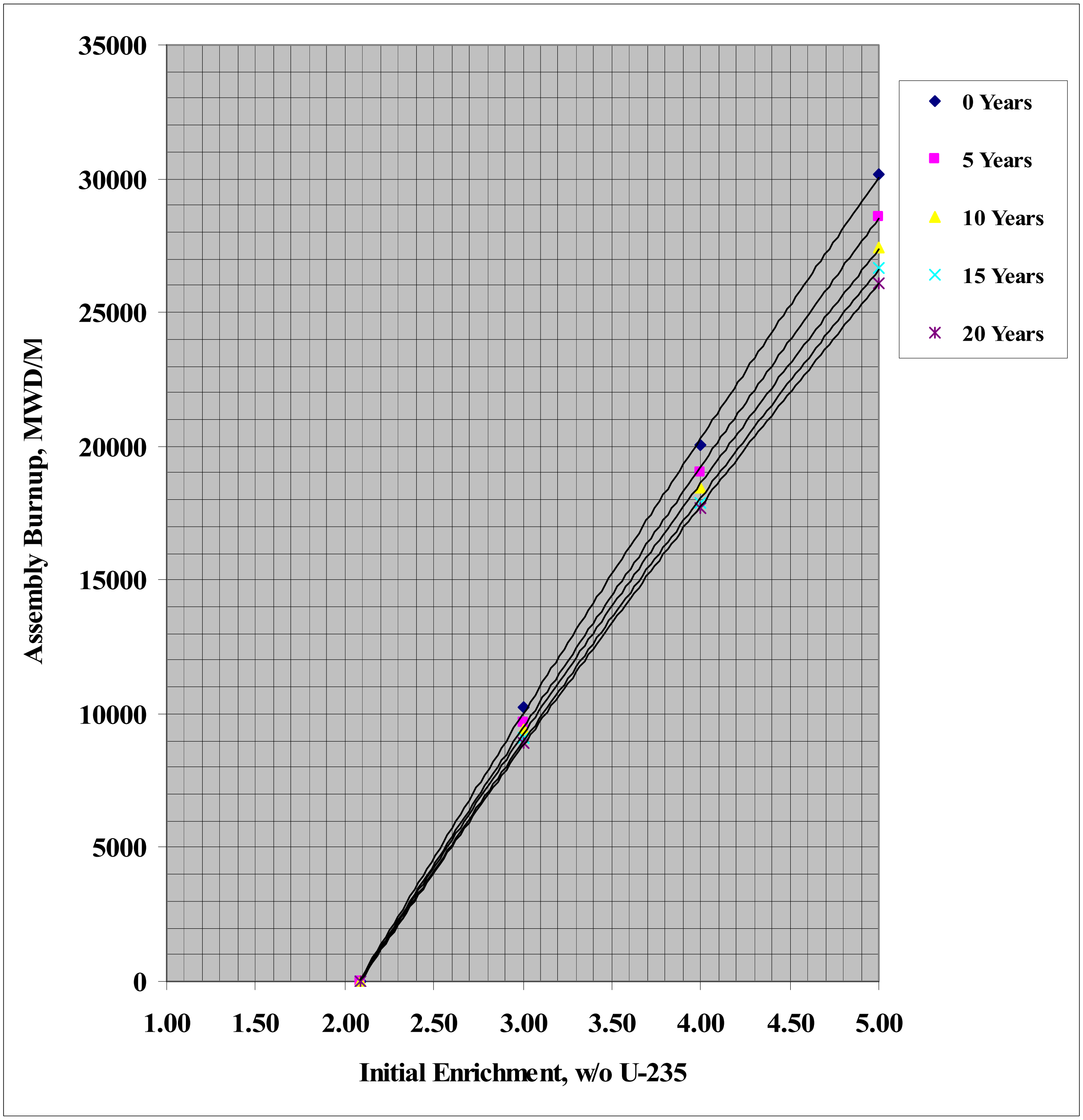

Figure 3.7.12-1

Minimum Fuel Assembly Burnup Versus Initial Enrichment for the Region 2 “All Cell” Storage Configurations

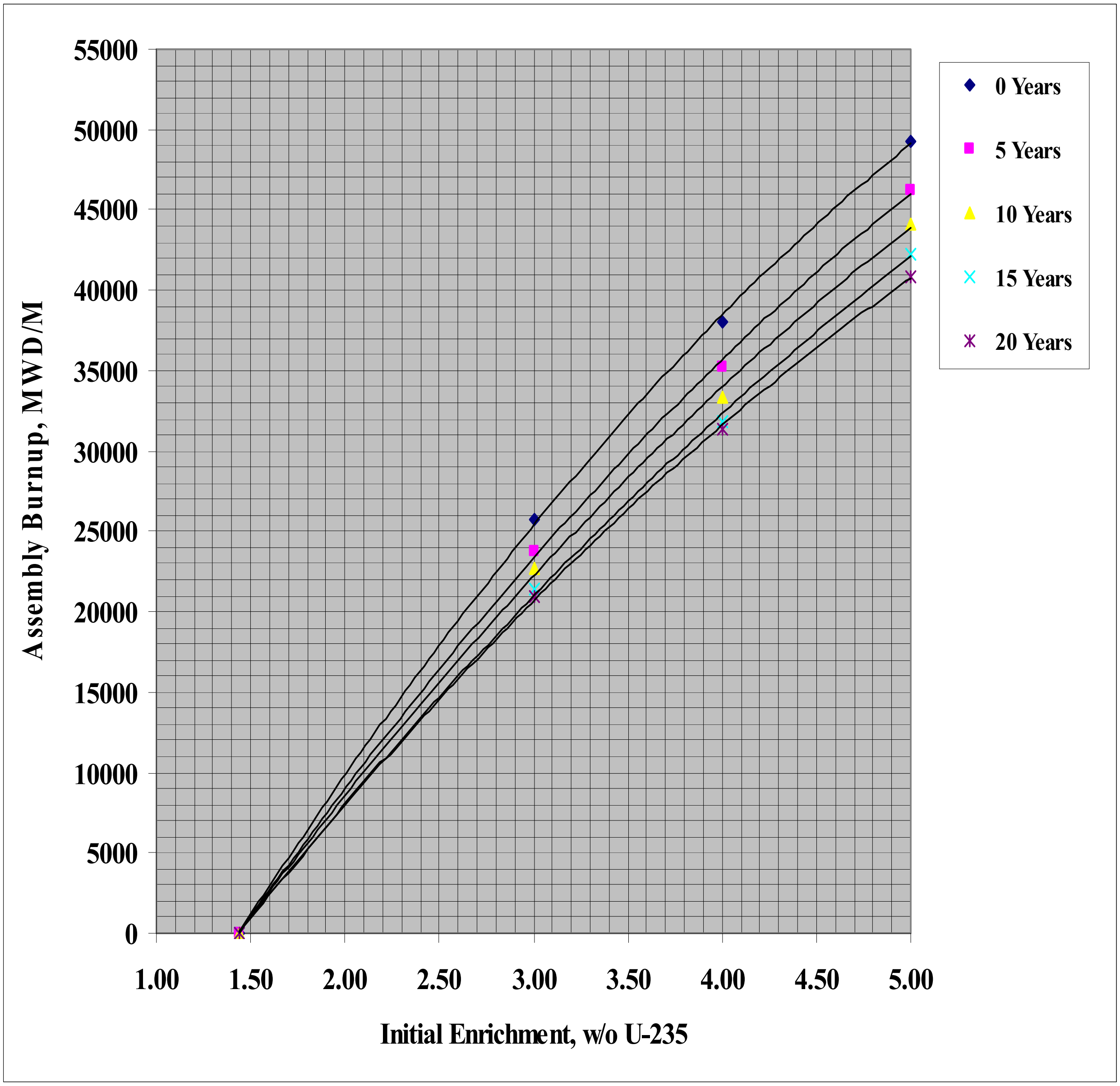

Figure 3.7.12-2

Minimum Fuel Assembly Burnup Versus Initial Enrichment for the Region 2 "1-out-of-4 5.0 Weight-Percent Fresh" Storage Configurations

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 DC Sources – Operating

LCO 3.8.1 The Division A, B, C, and D Class 1E DC power subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more battery chargers in one division inoperable.	A.1	Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	6 hours
	AND		
	A.2	Verify battery float current ≤ 2amps.	Once per 24 hours
	AND		
	A.3	Restore battery charger(s) to OPERABLE status.	7 days
B. One or more battery chargers in two divisions inoperable.	B.1	Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	2 hours
	AND		
	B.2	Verify battery float current ≤ 2amps.	Once per 24 hours
	AND		
	B.3	Restore battery charger(s) to OPERABLE status.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more batteries in one division inoperable.	C.1 Restore batteries to OPERABLE status.	6 hours
D. One or more batteries in two divisions inoperable.	D.1 Restore batteries to OPERABLE status.	2 hours
E. One DC electrical power subsystem inoperable for reasons other than Condition A or C.	E.1 Restore DC electrical power subsystem to OPERABLE status.	6 hours
F. Two DC electrical power subsystems inoperable for reasons other than B or D.	F.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours
G. Required Action and associated Completion Time not met.	G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	7 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.2	Verify each battery charger supplies ≥ 200 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours. <u>OR</u> Verify each 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
SR 3.8.1.3	--- **- NOTES -** 1. The modified performance discharge test in SR 3.8.7.6 may be performed in lieu of SR 3.8.1.3. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4 unless the spare battery is connected to replace the battery being tested. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. --- 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

3.8.2 DC Sources – Shutdown

LCO 3.8.2 DC electrical power subsystems shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.6, "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 DC electrical power subsystems inoperable.	A.1 Declare affected required features inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.4 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.2.5 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	-- **- NOTE -** The following SRs are not required to be performed: SR 3.8.1.2 and SR 3.8.1.3. -- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.1.1 SR 3.8.1.2 SR 3.8.1.3	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Inverters – Operating

LCO 3.8.3 The Division A, B, C, and D inverters (Divisions A and D, one each and Divisions B and C two each; six total) shall be OPERABLE.

- NOTES -

One inverter may be disconnected from its associated DC bus for ≤ 72 hours to perform an equalizing charge on its associated battery, providing:

1. The associated instrument and control bus is energized from its Class 1E constant voltage source transformer; and

2. All other AC instrument and control buses are energized from their associated OPERABLE inverters.

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 3.8.5 “Distribution Systems – Operating” with any instrument and control 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.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify correct inverter voltage, frequency, and alignment to required AC instrument and control buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 Inverters – Shutdown

LCO 3.8.4 Inverters shall be OPERABLE to support the onsite Class 1E power distribution subsystems required by LCO 3.8.6, "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 features inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.4 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.2.5 Initiate action to restore required inverters to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify correct inverter voltage, frequency, and alignments to required AC instrument and control buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 Distribution Systems – Operating

LCO 3.8.5 The Division A, B, C, and D AC instrument and control bus and DC electrical power distribution subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Division AC instrument and control bus inoperable.	A.1 Restore AC instrument and control bus to OPERABLE status.	6 hours <u>AND</u> 12 hours from discovery of failure to meet the LCO
B. One Division DC electrical power distribution subsystem inoperable.	B.1 Restore DC electrical power distribution subsystem to OPERABLE status.	6 hours <u>AND</u> 12 hours from discovery of failure to meet the LCO
C. Two Divisions AC instrument and control bus inoperable.	C.1 Restore AC instrument and control bus to OPERABLE status.	2 hours <u>AND</u> 16 hours from discovery of failure to meet the LCO.
D. Two Divisions DC electrical power distribution subsystem inoperable.	D.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours <u>AND</u> 16 hours from discovery of failure to meet the LCO.

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 5.	6 hours 36 hours
F. Two Divisions with inoperable distribution subsystems that result in a loss of safety function.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	Verify correct breaker and switch alignments and voltage to required DC and AC instrument and control bus electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Distribution Systems – Shutdown

LCO 3.8.6 The necessary portions of DC and AC instrument and control bus 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 DC or AC instrument and control bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required features inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.4 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.5 Initiate actions to restore required DC and AC instrument and control bus electrical power distribution subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify correct breaker and switch alignments and voltage to required DC and AC instrument and control bus electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Battery Parameters

LCO 3.8.7 Battery Parameters for Division A, B, C, and D batteries shall be within limits.

APPLICABILITY: When associated DC electrical power sources are required to be OPERABLE.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more batteries in one division with one or more battery cells float voltage < 2.07 V.	A.1	Perform SR 3.8.1.1.	2 hours
	AND		
	A.2	Perform SR 3.8.7.1.	2 hours
	AND		
	A.3	Restore affected cell voltage ≥ 2.07 V.	24 hours
B. One or more batteries in one division with float current > 2 amps.	B.1	Perform SR 3.8.1.1.	2 hours
	AND		
	B.2	Restore battery float current to ≤ 2 amps.	24 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 or more batteries in one division 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 or more batteries in one division 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 or more batteries in two or more divisions with battery parameters not within limits.	E.1	Restore battery parameters for batteries in three divisions to within limits.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time not met. <u>OR</u> One or more batteries in one division 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 3.8.7.1	--- **- NOTE -** Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.1.1. --- Verify each battery float current is ≤ 2 amps.	7 days
SR 3.8.7.2	Verify each battery pilot cell voltage is ≥ 2.07 V.	31 days
SR 3.8.7.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	31 days
SR 3.8.7.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	31 days
SR 3.8.7.5	Verify each battery connected cell voltage is ≥ 2.07 V.	92 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.7.6	-- **- NOTE-** This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, 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 <u>AND</u> 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentration of the Reactor Coolant System (RCS), the fuel transfer canal, and the refueling cavity shall be maintained within the limit specified in COLR.

APPLICABILITY: MODE 6.

- NOTE -

Only applicable to the fuel transfer canal and the refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate actions to restore boron concentration to within limits.	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 Unborated Water Source Flow Paths

LCO 3.9.2 Each unborated water source flow path shall be isolated.

APPLICABILITY: MODE 6.

ACTIONS

- NOTE -
Separate condition entry is allowed for each unborated water source flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------------------------- **- NOTE -** Required Action A.3 must be completed whenever Condition A is entered. ------------------------------- One or more flow paths not isolated.	A.1 Suspend CORE ALTERATIONS. AND A.2 Initiate actions to isolate flow paths. AND A.3 Perform SR 3.9.1.1, (boron concentration verification).	Immediately Immediately 4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Verify each unborated water source flow path is isolated by at least one valve secured in the closed position.	31 days

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 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 CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	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, (Boron Concentration Verification).	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Perform a CHANNEL CHECK.	12 hours
SR 3.9.3.2	-- **- NOTE-** Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	24 months

3.9 REFUELING OPERATIONS

3.9.4 Refueling Cavity Water Level

LCO 3.9.4 Refueling Cavity Water Level shall be maintained ≥ 23 ft above the top of the reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

- NOTE -

LCO 3.0.8 is not applicable.

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 3.9.4.1	Verify that refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

3.9 REFUELING OPERATIONS

3.9.5 Containment Penetrations

LCO 3.9.5 The containment penetrations shall be in the following status:

a. The equipment hatches closed and held in place by four bolts or, if open, the containment air filtration system (VFS) shall be OPERABLE and operating;

b. One door in each air lock closed or, if open, the VFS shall be OPERABLE and operating;

c. The containment spare penetrations closed or, if open, the VFS shall be OPERABLE and operating;

d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

   1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or

   2. Capable of being closed by an OPERABLE Containment Isolation signal.

- NOTE -

Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

- NOTE -

LCO 3.0.8 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. LCO not met.	A.1	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.5.2	-- **- NOTE-** Not required to be met for containment purge and exhaust valve(s) in penetrations closed to comply with LCO 3.9.5.d.1. -- Verify each required containment purge and exhaust valve actuates to the isolation position on a manual actuation signal.	In accordance with the Inservice Test Program
SR 3.9.5.3	Verify the VFS can maintain a negative pressure ($\leq$ -0.125 inches water gauge relative to outside atmospheric pressure) in the area enclosed by the containment and alternate barrier.	24 months
SR 3.9.5.4	Operate each VFS train for $\geq$ 10 continuous hours with the heaters operating.	Within 31 days prior to fuel movement or CORE ALTERATIONS

3.9 REFUELING OPERATIONS

3.9.6 Containment Air Filtration System (VFS)

LCO 3.9.6 One VFS exhaust subsystem shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel building.

ACTIONS

- NOTE -

LCOs 3.0.3 and 3.0.8 are not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required VFS exhaust subsystem inoperable.	A.1 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Operate each VFS exhaust subsystem for $\geq$ 10 continuous hours with the heaters operating.	Within 31 days prior to fuel movement
SR 3.9.6.2	Verify the VAS fuel handling area subsystem aligns to the VFS exhaust subsystem on an actual or simulated actuation signal.	24 months
SR 3.9.6.3	Verify one VFS exhaust subsystem can maintain a negative pressure ($\leq$ -0.125 inches water gauge relative to outside atmospheric pressure) in the fuel handling area.	24 months

3.9 REFUELING OPERATIONS

3.9.7 Decay Time

LCO 3.9.7 The reactor shall be subcritical for ≥ 48 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTIONS

- NOTE -

LCO 3.0.8 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical < 48 hours.	A.1 Suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify that the reactor has been subcritical for ≥ 48 hours by verification of the date and time of subcriticality.	Prior to movement of irradiated fuel in the reactor vessel

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4.0 DESIGN FEATURES

4.1 Site

The site for the Levy Nuclear Plant (LNP) is located in the southern part of Levy County, Florida, east of U.S. Highway 19/98 (SR55) and near the cities and towns of Inglis, Yankeetown, and Crystal River.

4.1.1 Site and Exclusion Boundaries

The Site Boundary is shown in Figure 4.1-1.

The Exclusion Area Boundary is shown in Figure 4.1-2.

4.1.2 Low Population Zone (LPZ)

The LPZ is defined by the 3 mile radius from the site center point as shown in Figure 4.1-1.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of fuel rods clad with a zirconium based alloy and containing an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium based 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 staff approved codes and methods and shown by tests or analyses to comply with 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 and Gray Rod Assemblies

The reactor core shall contain 53 Rod Cluster Control Assemblies (RCCAs), each with 24 rodlets/RCCA. The RCCA absorber material shall be silver indium cadmium as approved by the NRC.

Additionally, there are 16 low worth Gray Rod Cluster Assemblies (GRCAs), with 24 rodlets/GRCA, which, in conjunction with the RCCAs, are used to augment MSHIM load follow operation.

4.0 DESIGN FEATURES

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 Section 9.1, “Fuel Storage and Handling.”

c. A nominal 10.90 inch center-to-center distance between fuel assemblies placed in Region 1, a nominal 9.028 inch center-to-center distance between fuel assemblies placed in Region 2 of the spent fuel storage racks, and a nominal 11.62 inch center-to-center distance between fuel assemblies placed in the Defective Fuel Cells.

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

e. Partially spent fuel assemblies meeting the initial enrichment, burnup, and decay time requirements of LCO 3.7.12, “Spent Fuel Pool Storage,” may be stored in Region 2 of Figure 4.3-1, and

f. New and spent fuel assemblies meeting the Figure 4.3-2 location-specific initial enrichment, burnup, and decay time requirements of LCO 3.7.12, “Spent Fuel Pool Storage,” may be stored in specified Region 2 locations.

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 Section 9.1, “Fuel Storage and Handling.”

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam which includes an allowance for uncertainties as described in Section 9.1, “Fuel Storage and Handling.”

d. A nominal 10.90 inch center-to-center distance between fuel assemblies placed in the new fuel storage racks.

4.0 DESIGN FEATURES

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below a minimum water depth of ≥ 23 ft above the surface of the fuel storage racks.

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 889 fuel assemblies.

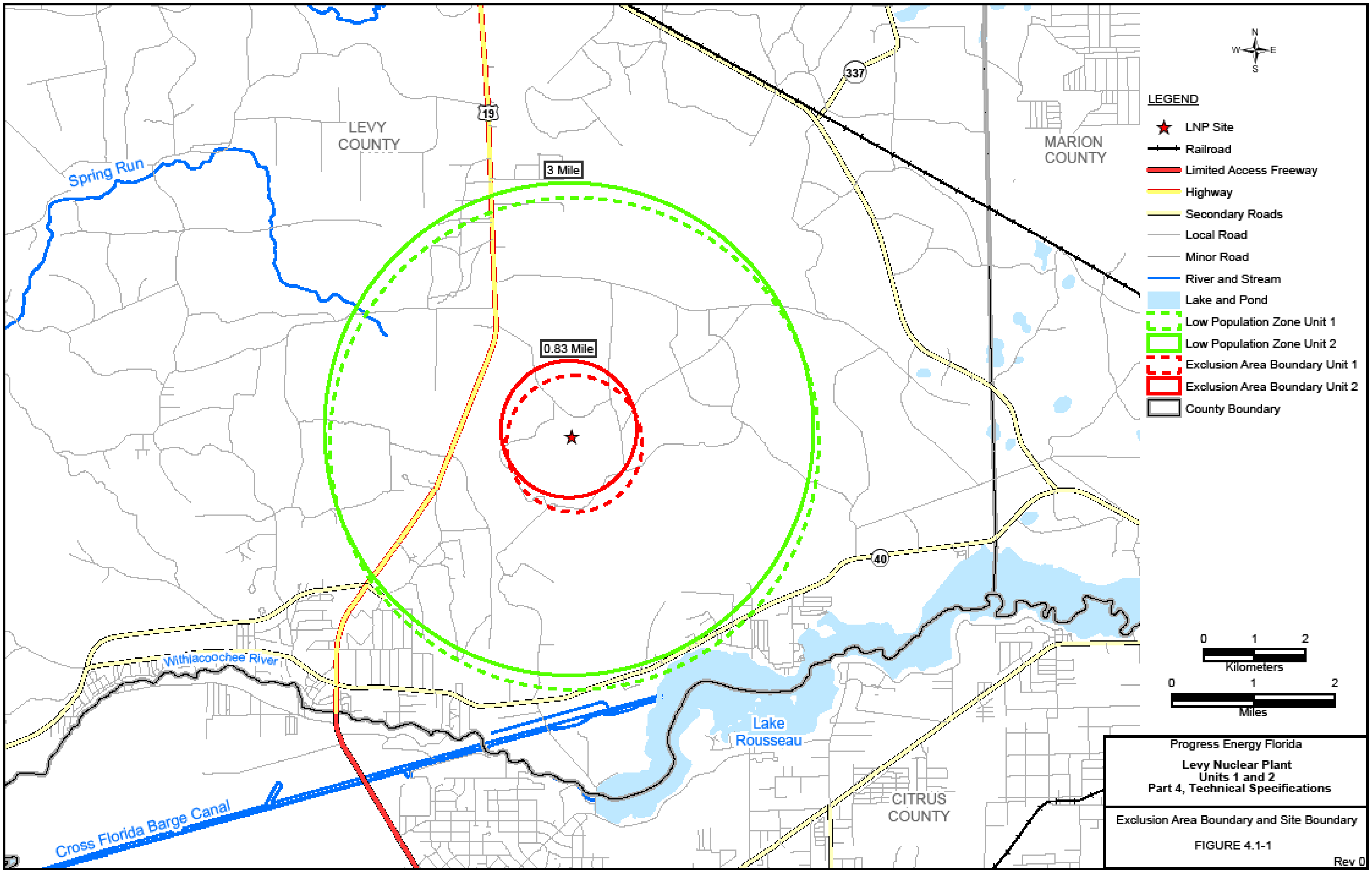


FIGURE 4.1-1 Low Population Zone and Site Boundary Map

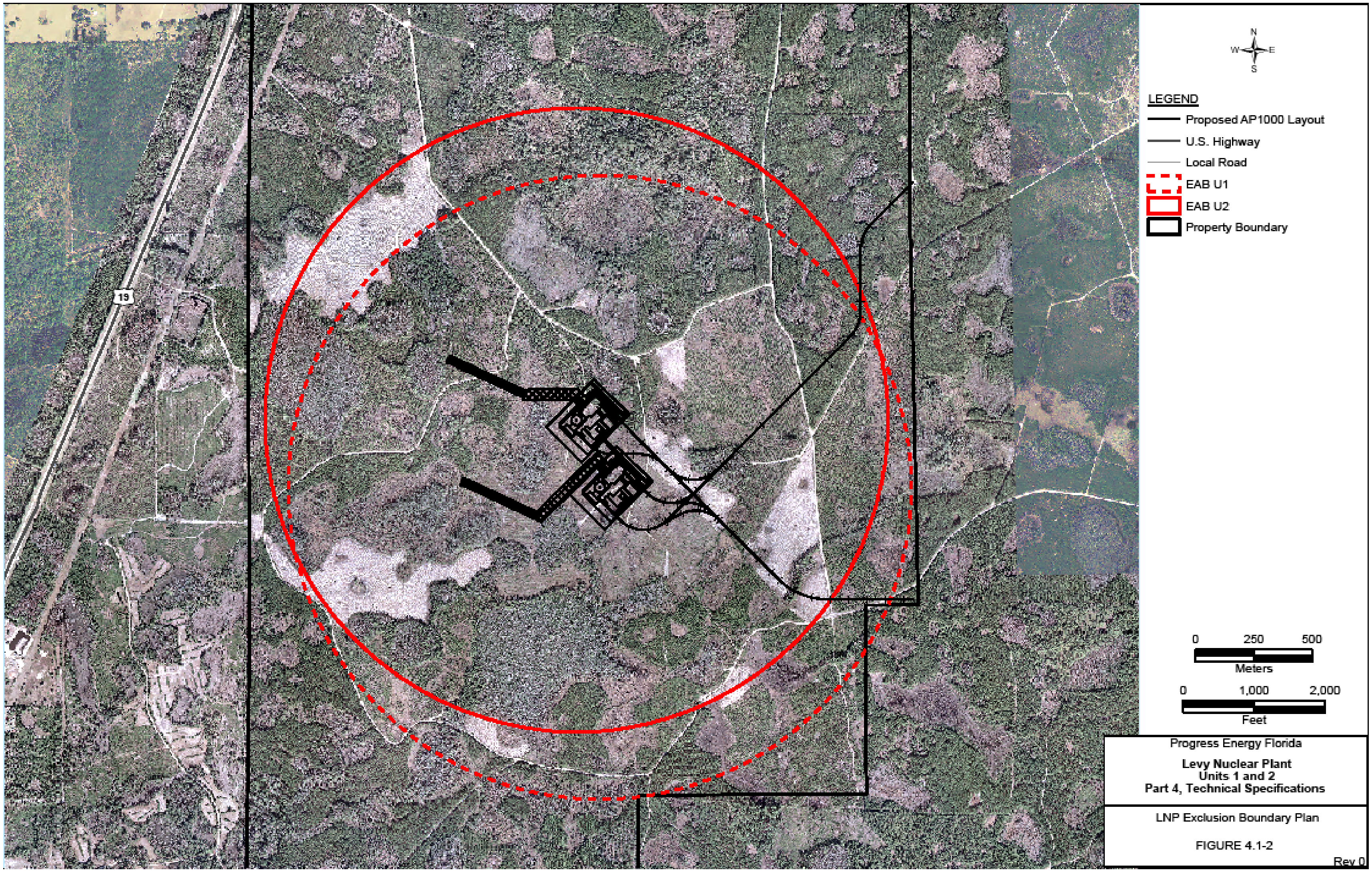
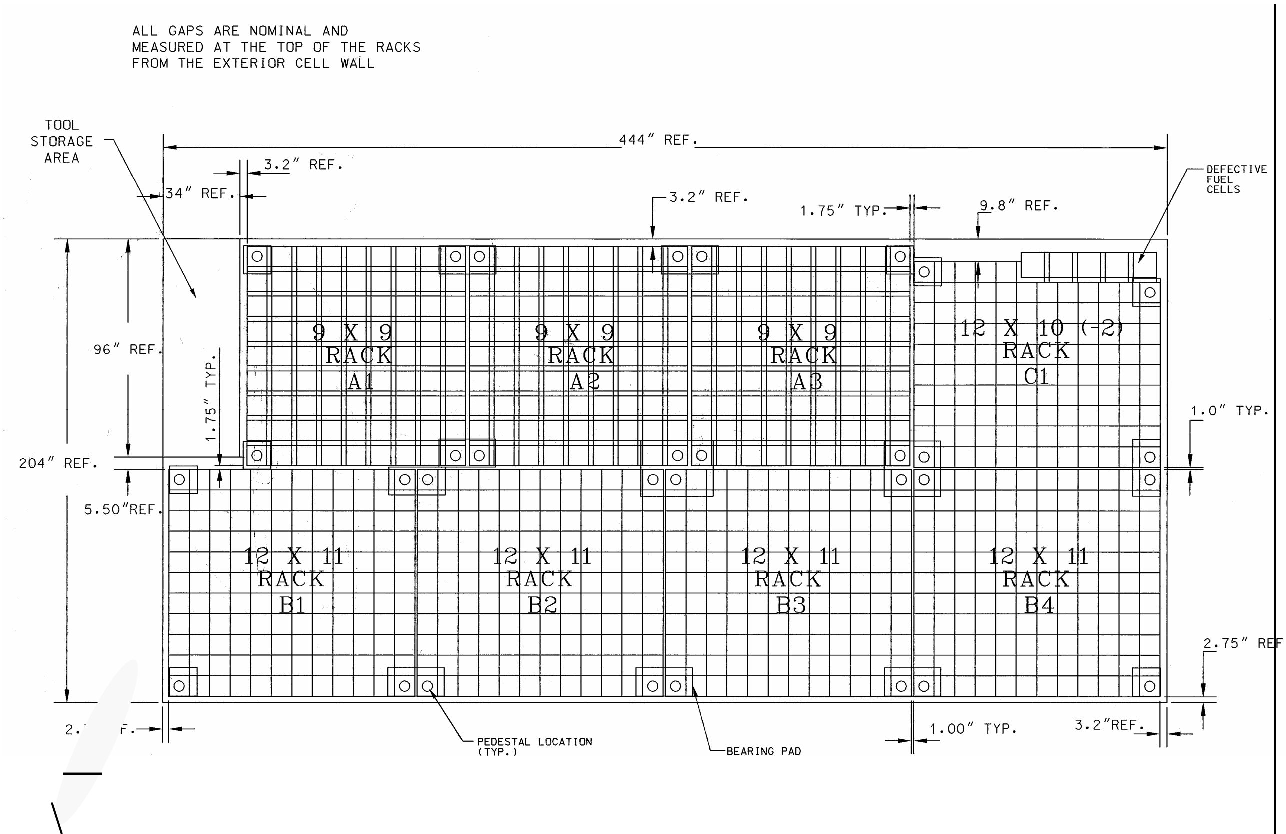


FIGURE 4.1-2 Exclusion Area Boundary



Region 1 (A1, A2, A3) – 243 locations

Region 2 (B1, B2, B3, B4, C1) – 641 locations

Defective Fuel Cells (DFCs) – 5 locations

Total Storage Locations – 889

Spent Fuel (equivalent to 1.361 w/o fresh fuel)	**New Fuel (5.0 w/o fresh fuel and all spent fuel)**
Spent Fuel (equivalent to 1.361 w/o fresh fuel)	**Spent Fuel (equivalent to 1.361 w/o fresh fuel)**

Figure 4.3-2

Region 2 “1-out-of-4 5.0 Weight-Percent Fresh” Fuel Configuration

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Plant General Manager shall be responsible for overall unit operations and shall delegate in writing the succession to this responsibility during his absence.

The Plant General 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 Nuclear Shift Manager (NSM) shall be responsible for the control room command function. During any absence of the NSM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the NSM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor 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;

b. The Plant General 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; and

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 operation 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 5.2.2.a and 5.2.2.e 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. The Manager – Operations or Manager – Shift Operations shall hold an SRO license.

e. 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 through the first refueling outage, the Regulatory Position C.1.b of Regulatory Guide 1.8, Revision 2, 1987, applies: cold license operator candidates meet the training elements defined in ANSI/ANS 3.1-1993 but are exempt from the experience requirements defined in ANSI/ANS 3.1-1993.

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 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 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 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the ODCM:

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

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. 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;

b. Shall become effective after the approval of the Plant General Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the changed portion of the 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 Radioactive Effluent Control 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 setpoints 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;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance 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 for radioactive materials in liquid effluents released form 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;

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 guidelines for 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 to areas 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

5.5 Programs and Manuals

5.5.2 Radioactive Effluent Control Program (continued)

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.

5.5.3 Inservice Testing Program

This program provides control for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies specified in the 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
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

5.5 Programs and Manuals

5.5.3 Inservice Testing Program (continued)

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

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.4 <u>Steam Generator (SG) Program</u>

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. 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.

5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

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. Leakage is not to exceed 150 gpd per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.7, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness 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 repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and 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 following SG replacement.

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.

5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

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.

5.5.5 <u>Secondary Water Chemistry Program</u>

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.

5.5.6 <u>Technical Specifications (TS) Bases Control Program</u>

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.

5.5 Programs and Manuals

5.5.6 Technical Specifications (TS) Bases Control Program (continued)

b. Licensees may make changes to Bases without prior NRC 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 (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).

5.5.7 Safety Function Determination Program (SFDP)

This program ensure loss of safety function is detected and appropriate action 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 supported system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirement of LCO 3.0.6. The SFDP shall contain the following:

a. 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;

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

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

d. Other appropriate limitations and remedial or compensatory actions.

5.5 Programs and Manuals

5.5.7 Safety Function Determination Program (continued)

A loss of safety function exists when, assuming no concurrent single failure, 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:

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

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

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

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.

5.5.8 Containment Leakage Rate Testing Program

a. A program shall be established to implement 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 approved exceptions.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 57.8 psig. The containment design pressure is 59 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.10% of primary containment air weight per day.

5.5 Programs and Manuals

5.5.8 Containment Leakage Rate Testing Program (continued)

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 Type 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.

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

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.9 System Level OPERABILITY Testing Program

The System Level OPERABILITY Testing Program provides requirements for performance tests of passive systems. The System Level Inservice Tests specified in Section 3.9.6 and Table 3.9-17 apply when specified by individual Surveillance Requirements.

a. The provisions of SR 3.0.2 are applicable to the test frequencies specified in Table 3.9-17 for performing system level OPERABILITY testing activities; and

b. The provisions of SR 3.0.3 are applicable to system level OPERABILITY testing activities.

5.5.10 Component Cyclic or Transient Limit

This program provides controls to track the Table 3.9-1A cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5 Programs and Manuals

5.5.11 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, “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 minimum established design limit.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 Occupational Radiation Exposure Report

- NOTE -

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), electronic dosimeter or film badge measurements. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year. The initial report shall be submitted by April 30 of the year following the initial criticality.

5.6.2 Annual Radiological Environmental Operating Report

- 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.

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

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.

5.6.3 Radioactive Effluent Release Report

- NOTE -

A single submittal may be made for a multiple unit station.

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 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

2.1.1, “Reactor Core SLs”
3.1.1, “SHUTDOWN MARGIN (SDM)”
3.1.3, “Moderator Temperature Coefficient”
3.1.5, “Shutdown Bank Insertion Limits”
3.1.6, “Control Bank Insertion Limits”
3.2.1, “Heat Flux Hot Channel Factor”

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

3.2.2, “Nuclear Enthalpy Rise Hot Channel Factor”
3.2.3, “AXIAL FLUX DIFFERENCE”
3.2.5, “OPDMS-monitored Power Distribution Parameters”
3.3.1, “Reactor Trip System (RTS) Instrumentation”
3.4.1, “RCS Pressure, Temperature, and DNB Limits”
3.9.1, “Boron Concentration”

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. WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985 (Westinghouse Proprietary) and WCAP-9273-NP-A (Non-Proprietary).

   (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limits, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - AXIAL FLUX DIFFERENCE, and 3.9.1 - Boron Concentration.)

2a. WCAP-8385, “Power Distribution Control and Load Following Procedures - Topical Report,” September 1974 (Westinghouse Proprietary) and WCAP-8403 (Non-Proprietary).

   (Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

2b. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

   (Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

   (Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

3. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification,” February 1994 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

   (Methodology for Specifications 3.2.3 - AXIAL FLUX DIFFERENCE (Relaxed Axial Offset Control) and 3.2.1 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for FQ Methodology).)

4. WCAP-12945-P-A, Volumes 1-5, “Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis,” Revision 2, March 1998 (Westinghouse Proprietary) and WCAP-14747 (Non-Proprietary).

   (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor.)

5. WCAP-12472-P-A, “BEACON Core Monitoring and Operations Support System,” August 1994, Addendum 1, May 1996 (Westinghouse Proprietary), and Addendum 2, March 2001 (Westinghouse Proprietary) and WCAP-12473-A (Non-Proprietary).

   (Methodology for Specification 3.2.5 - OPDMS - Monitored Power Distribution Parameters.)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Passive Core Cooling Systems 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.

5.6.6 <u>Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)</u>

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   3.4.3, “RCS Pressure and Temperature (P/T) Limits”
   3.4.14, “Low Temperature Overpressure Protection (LTOP) System”

5.6 Reporting Requirements

5.6.6 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

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 document:

WCAP-14040-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves." (Limits for LCO 3.4.3 and LCO 3.4.14).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for any revision or supplement thereto.

5.6.7 Post Accident Monitoring Report

When a report is required by Condition B of LCO 3.3.3, "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.

5.6.8 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 5.5.4, "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 in-situ testing, and

h. The effective 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 or group entering such an area shall possess:

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

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, or

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

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)

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

   (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 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or 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>

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 or gate 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.

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 or 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)

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.

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 group entering such an area shall possess:

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, or

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.

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 or 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)

4. In those cases where options (2) and (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.

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.

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core Safety Limits (SLs)

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not to be 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 the fuel centerline temperature stays below the melting temperature.

The restriction of this SL prevents overheating of the fuel and cladding, as well as possible cladding perforation, that 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 and Safety Monitoring System (PMS) and steam generator safety valves 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 Reactor Trip System (RTS) setpoints (Ref. 2), 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, ΔI, 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 PMS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RTS setpoints. 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 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 cold leg temperature for which the minimum DNBR is not less than the safety analysis 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 RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip 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 the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS cold leg temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies 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 steam generator 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 reactor trip functions are specified in LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.” 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 SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

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. Section 7.2, “Reactor Trip.”

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 coolant pressure 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 2500 psia (2485 psig). 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 American Society of Mechanical Engineers (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.

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressurizer pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety 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 with loss of feedwater flow, without a direct reactor trip. During the transient, no control actions are assumed except that the

BASES

APPLICABLE SAFETY ANALYSES (continued)

safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressurizer pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressurizer pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. RCS depressurization valves;

b. Steam line relief valves (SG PORVs);

c. Turbine Bypass System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel, piping, valves, and fittings under the ASME Code, Section III, is 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2733.5 psig.

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 since the reactor vessel 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 abnormal radioactive releases.

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 50.34.

5. Section 7.2, “Reactor Trip.”

B 3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.8 establish the general requirements applicable to all Specifications 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 requirements 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 the 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.

BASES

LCO 3.0.2 (continued)

Completing the Required Actions is not required when an LCO is met, or is no longer applicable, unless otherwise stated in the individual Specifications.

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. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions could exist which 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

LCO 3.0.3 (continued)

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.

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 into 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. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant 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.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition was initially entered and not from the time LCO 3.0.3 is exited.

LCO 3.0.3 (continued)

The time limits of Specification 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 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 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. 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. In MODES 5 and 6, LCO 3.0.8 provides actions for Conditions not covered in other Specifications.

Exceptions to 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 3.7.5, Spent Fuel Pool Water Level. This Specification has an Applicability of “At all times.” Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.5 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 3.7.5 of “Suspend movement of irradiated fuel assemblies in the spent fuel 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 precludes placing the unit in a MODE or other specified condition stated that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

LCO 3.0.4 (continued)

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

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. 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 results from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. These exceptions allow 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. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4 or 5, MODE 2 from MODE 3 or 4 or 5, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or 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.

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, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or

LCO 3.0.4 (continued)

SR 3.0.4 for those Surveillances that do not have to be performed due to the associated 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 of 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 Surveillance Requirements 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 SRs.

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 an SR 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 an SR on another channel in the same trip system.

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

LCO 3.0.6 (continued)

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 5.5.7, "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. 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

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LCO 3.0.6 (continued)

safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety functions 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.

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 addresses 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 support 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 LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performance 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.

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

BASES

LCO 3.0.7 (continued)

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 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.

This Specification delineates the requirements 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 3.0.8, 1 hour is allowed to prepare for an orderly plan of action which optimizes plant safety and equipment restoration. The Shutdown Safety Status Trees provide a systematic method to explicitly determine the status of the plant during shutdown conditions, after entering MODE 5. A set of plant parameters is monitored and if any parameter is outside of its defined limits, a transition is made to the Shutdown Emergency Response Guidelines. These guidelines provide preplanned actions for addressing parameters outside defined limits.

Examples of the required end states specified for inoperable passive systems while in MODES 5 and 6 are provided in Table B 3.0-1, Passive Systems Shutdown MODE Matrix. These requirements are specified in the individual Specifications. The required end states specified for passive systems, when the unit is in MODE 5 or 6, are selected to ensure that the initial conditions and system and equipment availabilities minimize the likelihood and consequences of potential shutdown events.

LCO 3.0.8 (continued)

ACTIONS required in accordance with LCO 3.0.8 may be terminated and LCO 3.0.8 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.

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 3.0.8 is exited.

In MODES 5 and 6, LCO 3.0.8 provides actions for Conditions not covered in other Specifications and for multiple concurrent Conditions for which conflicting actions are specified.

As an example of the application of LCO 3.0.8, see column 2 of Table B 3.0-1, Passive Systems Shutdown MODE Matrix, for the core makeup tank. This example assumes that the plant is initially in MODE 5 with the RCS pressure boundary intact. In this plant condition, LCO 3.5.3 requires one core makeup tank to be OPERABLE. The table shows the required end state established by the Required Actions of TS 3.5.3 in the event that the core makeup tank cannot be restored to OPERABLE status.

For this initial plant shutdown condition with no OPERABLE core makeup tanks, four conditions are identified in TS 3.5.3, with associated Required Actions and Completion Times. If Conditions A, B, and C cannot be completed within the required Completion Times, then Condition D requires immediately initiating action to place the plant in MODE 5 with the RCS pressure boundary open, and with pressurizer level greater than 20 percent.

LCO 3.0.8 would apply if actions could not immediately be initiated to open the RCS pressure boundary. In this situation, in parallel with the TS 3.5.3 actions to continue to open the RCS pressure boundary, LCO 3.0.8 requires the operators to take actions to restore one core makeup tank to OPERABLE status, and to monitor the Safety System Shutdown Monitoring Trees.

The Shutdown Status Trees monitor seven key RCS parameters and direct the operators to one of six shutdown ERGs in the event that any of the parameters are outside of allowable limits. The shutdown ERGs

BASES

LCO 3.0.8 (continued)

identify actions to be taken by the operators to satisfy the critical safety functions for the plant in the shutdown condition, using plant equipment available in this shutdown condition. LCO 3.0.8 monitoring would continue to be required until one core makeup tank is restored to OPERABLE status or the Required Actions for Condition D can be satisfied. In this case, once the RCS pressure boundary is open as required by Condition D, LCO 3.0.8 would be exited.

Table B 3.0-1 (page 1 of 1)
Passive Systems Shutdown MODE Matrix

LCO Applicability	Automatic Depressurization System	Core Makeup Tank	Passive RHR	IRWST	Containment	Containment Cooling[(1)]
MODE 5 RCS pressure boundary intact	9 of 10 paths OPERABLE All paths closed LCO 3.4.12	One CMT OPERABLE LCO 3.5.3	System OPERABLE LCO 3.5.5	One injection flow path and one recirculation sump flow path OPERABLE LCO 3.5.7	Closure capability LCO 3.6.8	Three water flow paths OPERABLE LCO 3.6.7
Required End State	MODE 5 RCS pressure boundary open, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary open, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary open, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level
MODE 5 RCS pressure boundary open or pressurizer level < 20%	Stages 1, 2, and 3 open 2 stage 4 valves OPERABLE LCO 3.4.13	None	None	One injection flow path and one recirculation sump flow path OPERABLE LCO 3.5.7	Closure capability LCO 3.6.8	Three water flow paths OPERABLE LCO 3.6.7
Required End State	MODE 5 RCS pressure boundary open, ≥ 20% pressurizer level			MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level	MODE 5 RCS pressure boundary intact, ≥ 20% pressurizer level
MODE 6 Upper internals in place	Stages 1, 2, and 3 open 2 stage 4 valves OPERABLE LCO 3.4.13	None	None	One injection flow path and one recirculation sump flow path OPERABLE LCO 3.5.8	Closure capability LCO 3.6.8	Three water flow paths OPERABLE LCO 3.6.7
Required End State	MODE 6 Upper internals removed			MODE 6 Refueling cavity full	MODE 6 Refueling cavity full	MODE 6 Refueling cavity full
MODE 6 Upper internals removed	None	None	None	One injection flow path and one recirc-ulation sump flow path OPERABLE LCO 3.5.8	Closure capability LCO 3.6.8	Three water flow paths OPERABLE LCO 3.6.7
Required End State				MODE 6 Refueling cavity full	MODE 6 Refueling cavity full	MODE 6 Refueling cavity full

(1) Containment cooling via PCS is not required when core decay heat < 9 MWt.

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 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 ensures 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.

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. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met in accordance with SR 3.0.2 prior to returning equipment to OPERABLE status.

SR 3.0.1 (continued)

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.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Actions 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. 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 remedial action, is

SR 3.0.2 (continued)

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 compliance 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. When a Surveillance with a Frequency based not on time intervals, but upon specified unit Conditions or operational 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.

SR 3.0.3 (continued)

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 plant risk (from delaying the Surveillance as well as any plant 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, NRC 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 plant 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.

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 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 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 component to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

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) apply to MODE or other specified condition changes.

The provisions of SR 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.

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 a 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 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

BASES

SR 3.0.4 (continued)

required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SR's annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or 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.

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 shutdown under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to assure 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 Rod Cluster Control Assemblies (RCCAs), assuming that the single rod cluster assembly 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 Plant Control System (PLS) 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 PLS, 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 rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM is calculated and monitored by the OPDMS and controlled by operating with RCCAs sufficiently withdrawn to meet the SDM requirement. When the OPDMS is inoperable, SDM control is ensured by operating within the limits of LCO 3.1.5 “Shutdown Bank Insertion Limits,” and LCO 3.1.6, “Control Bank Insertion Limits.” When the unit is in the shutdown and refueling modes, the SDM requirements are met by adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analyses (Ref. 2) establish an SDM that ensures that 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, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departures from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accidents for the SDM requirements are based on a main steam line break (SLB) and inadvertent opening of a steam generator (SG) relief or safety valve, as described in the accident analyses (Ref. 2). The increased steam flow in the main steam system causes an increased energy removal from the affected 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 (MTC), this cooldown causes an increase in core reactivity. 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 SLB or opening of an SG relief or safety valve, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and the THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting SLB and inadvertent opening of an SG relief or safety valve transients, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition;

APPLICABLE SAFETY ANALYSES (continued)

c. Rod ejection;

d. Inadvertent operation of Passive Residual Heat Removal Heat Exchanger (PRHR HX).

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 when critical boron concentrations are highest.

The uncontrolled rod withdrawal transient is terminated by a high neutron flux trip. Power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

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.

The inadvertent actuation of the PRHR HX causes an RCS temperature reduction from an initial injection of relatively cold water and the continued cooling of the RCS by PRHR. In the presence of a negative moderator temperature coefficient, the RCS temperature reduction causes an increase in core reactivity. Safety injection on the low cold leg temperature or low pressurizer pressure signals actuate the core makeup tank (CMT) and bring the plant to a stable condition.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the main control room, SDM is considered an initial condition process variable because it is periodically monitored to provide assurance that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through calculations by the Online Power Distribution Monitoring System (OPDMS) and RCCA positioning and through the soluble boron concentration.

BASES

LCO (continued)

The SLB and the boron dilution accidents (Ref. 2) are the most limiting analyses that establish the SDM value of the LCO. For SLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 50.34 limits (Ref. 3). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for automatic action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$, and 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.”

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 concentrated solution. The operator should begin boration with the best source available for the plant conditions.

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 hot shutdown conditions when boron concentration is highest at 1502 ppm. Assuming that a value of 1.0% $\Delta k/k$ must be recovered and the boration flow rate is 100 gpm, it is possible to increase the boron concentration of the RCS by 111 ppm in approximately 21 minutes utilizing boric acid solution having a concentration of 4375 ppm. If a boron worth of 9 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1.0% $\Delta k/k$. These boration parameters of 100 gpm and 4375 ppm represent typical values and are provided for the purpose of offering a specific example.

BASES

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 an RCCA is known to be untrippable, however, SDM verification must account for the worth of both the untrippable RCCA as well as another RCCA of maximum worth.

In MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering at least the listed reactivity effects:

a. RCS boron concentration;

b. RCCA and GRCA 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).

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.

2. Chapter 15, “Accident Analysis.”

3. 10 CFR 50.34.

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. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity and could potentially result in a loss of SDM 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 rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculation 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 a negative moderator temperature coefficient, the

BASES

BACKGROUND (continued)

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.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to compensate reactivity and maintain 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. Certain accident evaluations (Ref. 2) are, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are sensitive to accurate predictions 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 provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analysis 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 a 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 calculation 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

APPLICABLE SAFETY ANALYSES (continued)

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.

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.

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(c)(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 Conditions of 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 DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta k/k$ has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% $\Delta k/k$ 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.

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 MODE 3, 4, and 5 because the reactor is shutdown and the reactivity balance is not changing.

BASES

APPLICABILITY (continued)

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 during the first startup following 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 DBA 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.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

BASES

ACTIONS (continued)

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, 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 the boration required by SR 3.1.1.1 would occur. 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 control rod 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 Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPDs) 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 EFPDs following the initial 60 EFPDs after entering MODE 1 is acceptable based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. Chapter 15, "Accident Analysis."

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 non-positive MTC over the 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 MTC is less than zero when THERMAL POWER is at 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 (burnable absorbers) to yield an MTC 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 designed to achieve high burnups that have changes to other characteristics 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 Chapter 15 accident and transient analyses (Ref. 2).

If the LCO limits are not met, the plant 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 RCS boron concentration associated with fuel burnup and burnable absorbers.

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.

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 least negative 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 heat-up must be evaluated when the MTC is least negative. Such accidents include the rod withdrawal transient from either zero (Ref. 2) 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 BOC or EOC. 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 the limiting time in cycle life. An EOC measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

BASES

APPLICABLE SAFETY ANALYSES (continued)

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

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 accident analysis during operation.

Assumptions made in safety analyses require that the MTC be more negative than a given upper limit and less negative than a given lower limit. The MTC is least negative near BOC; this upper bound must not be exceeded. This maximum upper limit occurs at 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 BOC limit and the EOC 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 assure 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

BASES

APPLICABILITY (continued)

MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES.

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 will be reduced. 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 placed in MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC which is less negative 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 EOC MTC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the EOC MTC limit is exceeded, the plant must be placed in a MODE or Condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 4 within 12 hours.

The allowed Completion Time is a reasonable time based on operating experience to reach the required MODE from full power operation in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most limiting MTC LCO. Meeting the limit prior to entering MODE 1 assures that the limit will also be met at higher power levels.

The BOC 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 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 for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to provide assurance that the LCO limit will be met at EOC when the 300 ppm Surveillance criterion is met.

SR 3.1.3.2 is modified by three Notes that include the following requirements:

a. The SR is not required to be performed until 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

b. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOC limit.

c. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.

2. Chapter 15, “Accident Analysis.”

3. WCAP 9273-NP-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the RCCAs is an initial assumption in all safety analyses which 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. Gray Rod Cluster Assemblies (GRCAs) are excluded from this LCO during the planned swap of the gray rod banks, with OPDMS operable. The swap of GRCA banks will be periodically necessary to prevent excessive burnup shadowing of fuel rods near the gray rod assemblies. The bank swap maneuver will purposefully misalign GRCAs from their bank for a short period of time. The exclusion from this LCO is acceptable due to SHUTDOWN MARGIN being calculated exclusive of GRCAs, the relative low worth of individual gray rod assemblies, the short time duration anticipated for the swap maneuver and with OPDMS operable, power peaking and xenon redistribution effects will be monitored and controlled.

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” (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 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 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) and GRCAs are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA or GRCA one step (approximately 5/8 inch) at a time but at varying rates (steps per minute) depending on the signal output from the Plant Control System (PLS).

BASES

BACKGROUND (continued)

The rod control assemblies 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 rod control assemblies that are electrically paralleled to step simultaneously. A bank of rod control assemblies consists of two groups that are moved in a staggered fashion, but always within one step of each other. The AP1000 design has seven control banks and four shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are part of the MSHIM (Mechanical Shim) Control System which utilizes two independently OPERABLE groups of control banks for control of reactivity and axial power distribution.

Certain control rods will be pre-selected for inclusion in the Rapid Power Reduction (RPR) system. The purpose of the RPR is to initiate a rapid decrease in the core power during load rejection transients.

Reactivity control is provided primarily by the M banks. The M Banks consist of several control banks operating with a fixed overlap. The bank worth and overlap are defined so as to minimize the impact on axial offset with control bank maneuvering and still retain the reactivity required to meet the desired load changes.

The axial power distribution control is provided by the AO Bank, a relatively high worth bank.

In order to avoid boron adjustment for load follow operation, gray rods are utilized.

There are 16 GRCAs in the AP1000, each composed of 24 rodlets mounted on a common RCCA spider. These have been subdivided into what has been termed as MA, MB, MC, and MD Banks with 4 GRCAs in each.

Each of the MA, MB, MC, and MD Banks has almost the same worth. The primary gray bank function is to provide additional reactivity during the transition periods. During base load operation, two of the gray banks may be fully inserted into the core. Each of the gray banks consists of a relatively low worth bank.

The MA, MB, MC, MD, M1 and M2 Banks function together with a single variable (i.e., criticality or temperature) driving these groups as if they are in one control group.

BASES

BACKGROUND (continued)

The control rods 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 rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half-accuracy. The DRPI System is capable of monitoring rod position within at least ± 12 steps with either full accuracy or half accuracy.

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment is that:

a. There 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 accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a

APPLICABLE SAFETY ANALYSES (continued)

reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at or above their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

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 assure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the OPDMS indicates margin to limits or, if the OPDMS is inoperable, the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The limits on shutdown or control rod alignments assure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY assure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 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 linear heating rates (LHR), or unacceptable SDMs, which may constitute initial conditions inconsistent with the safety analysis.

The LCO is modified by a Note to relax the rod alignment limit on gray rods during GRCA swap operations. This operation which occurs frequently throughout the fuel cycle would normally violate the LCO.

APPLICABILITY

The requirements on RCCA OPERABILITY and 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 rods 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 rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has 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.

BASES

ACTIONS

A.1.1 and A.1.2

In this situation, SDM verification must include the worth of the untrippable rod as well as a rod of maximum worth.

When one or more rods 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 to determine SDM and, if necessary, to initiate boration to restore SDM.

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, 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 without challenging plant systems.

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable. With the OPDMS OPERABLE adverse peaking factors resulting from the misalignment can be detected. If the rod can be realigned within the Completion Time of 8 hours adverse burnup shadowing in the location of the misaligned rod can be avoided. With the OPDMS inoperable xenon redistribution can potentially cause adverse peaking factors which may not be detected. However, if the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, “Shutdown Bank Insertion Limit,” and LCO 3.1.6, “Control Bank Insertion Limits.” The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

BASES

ACTIONS (continued)

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified within limit or boration must be initiated to restore SDM within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank M2 to a rod that is misaligned 15 steps from the top of the core could require insertion of the M1 bank to maintain overlap limits.

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 to determine the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible. A note has been added indicating that Required Actions B.2.4 and B.2.5, F_Q and $F_{\Delta H}$ verification, are only required when the OPDMS is inoperable and therefore unavailable to continuously monitor the core power distribution.

Reduction of power to 75% of RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 3). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Protection and Safety Monitoring System.

When a rod 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.

Online monitoring of core power distribution by the OPDMS, or verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits when the OPDMS is inoperable, ensures that current operation at 75% of RTP with a rod misaligned is not resulting in power distributions which may invalidate safety analysis assumptions at full power. The Completion Time of

BASES

ACTIONS (continued)

72 hours allows sufficient time to restore OPDMS operable or to obtain and analyze offline flux maps of the core power distribution using the incore detector system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

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 a Design Basis Accident (DBA) 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 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 Times, 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 at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power condition in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod 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 of 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 CVS makeup pumps. Boration will continue until the required SDM is restored.

BASES

ACTIONS (continued)

D.2

If more than one rod 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 rods must be brought to within the alignment limits within 6 hours or 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 at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect that a rod is beginning to deviate from its expected position. The specified Frequency takes into account other rod position information that is continuously available to the operator in the main control room so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 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 rods. Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken. GRCA are excluded from this Surveillance because they are not considered in the calculation of SDM in MODES 1 and 2.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod 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 rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod 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 conservative conditions. GRCA are excluded from this Surveillance because they are not considered in the calculation of SDM in MODES 1 and 2.

This Surveillance is performed during a plant outage 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, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. Chapter 15, “Accident Analysis.”

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in the safety analyses which assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth 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 rod cluster control assemblies (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. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. The AP1000 design has seven control banks and four shutdown banks. See LCO 3.1.4, “Rod Group Alignment Limits,” for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, “Rod Position Indication,” for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Plant Control System (PLS), 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. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks

BASES

BACKGROUND (continued)

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. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks exclusive of the GRCAs), 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 full power conditions at rated temperature to zero power, and to maintain the required SDM at the rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown bank rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

a. There be 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 satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This in conjunction with LCO 3.1.6, “Control Bank Insertion Limits,” 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.

APPLICABILITY

The shutdown banks must be within their insertion limits with the reactor in MODE 1 and MODE 2. The LCO is not applicable if OPDMS is OPERABLE since OPDMS 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 MODE 3, 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 that 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 is not within insertion limits, 2 hours are 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.

BASES

ACTIONS (continued)

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits 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 main control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hours Frequency takes into account other information available in the main 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.

2. 10 CFR 50.46.

3. Chapter 15, “Accident Analysis.”

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in the safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available 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 rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks, gray rod cluster assemblies (GRCAs) are limited to control banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs or GRCAs 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 1 step of each other. The AP1000 design has seven control banks and four shutdown banks. See LCO 3.1.4, “Rod Group Alignment Limits,” for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, “Rod Position Indication,” for position indication requirements.

The control bank insertion sequence and overlap limits are specified in the COLR. The control banks are required to be at or above the insertion limit lines.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Plant Control System (PLS), but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.1.4, “Rod Group Alignment Limits,” LCO 3.1.5, “Shutdown Bank Insertion Limits,”

BACKGROUND (continued)

LCO 3.1.6, "Control Bank Insertion Limits," and LCO 3.2.5, "OPDMS – Monitored Powered Distribution Parameters," when the OPDMS is OPERABLE, or LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," when the OPDMS is inoperable, provide limits on control component operation and on monitored process variables which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits and power distribution limits 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 rod ejection accident, and the shutdown and control bank insertion limits assure 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 loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

   1. specified fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin which assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed QPTR 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 worth.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 3).

The insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) in that they are initial conditions assumed in the safety analysis.

LCO

The limits on control banks 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 rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. The LCO is not applicable if OPDMS is OPERABLE since OPDMS ensures that the limits are met. These limits must be maintained since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions.

Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements are modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. 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.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2, 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.

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 overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain outside the insertion limits for an extended period of time.

C.1

If Required Actions A.1 and A.2, or B.1 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 condition 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The estimated critical position (ECP) 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 ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP 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 ECP calculation with other startup activities.

SR 3.1.6.2

Verification of the control banks insertion limits at a Frequency of 12 hours is sufficient to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.

2. 10 CFR 50.46.

3. Chapter 15, "Accident Analysis."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Rod 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 (AOOs), 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 the safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the RCCA misalignment safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

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 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 control banks and shutdown banks. Each 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 systems: the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

BASES

BACKGROUND (continued)

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group receive the same signal to move and should, therefore, be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will function at half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod 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,” LCO 3.1.6, “Control Bank Insertion Limits”). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, “Rod Group Alignment Limits”). Control rod positions are continuously monitored to provide operators with information that assures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The control rod position indicators monitor control rod position, which is an initial condition of the accident.

BASES

LCO

LCO 3.1.7 specifies that one DRPI System and one Bank Demand Position Indication System be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

a. The DRPI System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, “Rod Group Alignment Limits”;

b. For the DRPI System there are no failed coils; and

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the DRPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System 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 given in LCO 3.1.4 in position indication for a single control rod ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements provide adequate assurance that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCOs 3.1.4, 3.1.5, and 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods has 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 (RCS).

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

When one DRPI channel per group fails, the position of the rod can still be determined by use of the On-line Power Distribution Monitoring System (OPDMS). Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Actions of B.1 or B.2 below are required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate to allow continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 2).

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 rod position determination by Required Action A.1 above.

B.1, B.2, B.3, and B.4

When more than one DRPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via incore detectors will minimize the potential for rod misalignment. The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition.

Monitoring and recording reactor coolant T_{avg} help assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

BASES

ACTIONS (continued)

The position of the rods may be determined indirectly by use of the incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SDM is within the limits provided in the COLR,

provided the nonindicating rods have not been moved. Verification of control rod position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with the shutdown without full rod position indication.

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved, the Required Action of C.1 and C.2 below is required.

<u>C.1 and C.2</u>

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the position was last determined, the Required Actions of A.1 and A.2 or B.1 are still appropriate but must be initiated promptly under Required Action C.1 to begin verifying that these rods are still properly positioned relative to their group positions.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

<u>D.1.1 and D.1.2</u>

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

ACTIONS (continued)

D.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 2). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions D.1.1 and D.1.2 or reduce power to ≤ 50% RTP.

E.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 at least MODE 3 within 6 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the DRPI agrees with the demand position within 12 steps provides assurance that the DRPI is operating correctly. Since the DRPI does not display the actual shutdown rod positions between 18 and 249 steps, only points within the indicated ranges are compared.

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 13.

2. Chapter 15, “Accident Analysis.”

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 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 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

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 TEST requirements for reload fuel cycles assure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TEST procedures are written and approved in accordance with established formats. The procedures include 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.

BACKGROUND (continued)

The typical PHYSICS TESTS performed for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration – Control Rods Withdrawn;

b. Control Rod Worth;

c. Isothermal Temperature Coefficient (ITC).

These tests are performed in MODE 2. 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 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 Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has four 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 calculated 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 while varying the reactor coolant boron concentration to maintain 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. The fourth method, Dynamic Rod

BASES

BACKGROUND (continued)

Worth Measurement (DRWM), moves each bank, individually, into the core to determine its worth. The bank is dynamically inserted into the core while data is acquired from the excore channel. While the bank is being withdrawn, the data is analyzed to determine the worth of the bank. This is repeated for each control and shutdown bank. Performance of this test will violate LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," or LCO 3.1.6, "Control Bank Insertion Limits."

c. The ITC Test measures the ITC of the reactor. This test is performed at HZP. The method is to vary the 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. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology report (Ref. 5). 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.

Chapter 14 defines requirements for initial testing of the facility, including low power PHYSICS TESTS. Sections 14.2.10.2 and 14.2.10.3 (Ref. 6) summarize the initial criticality and low power tests.

Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for the 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 3.1.3 "Moderator Temperature Coefficient (MTC),"
LCO 3.1.4 "Rod Group Alignment Limits,"
LCO 3.1.5 "Shutdown Bank Insertion Limit,"

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.1.6 "Control Bank Insertion Limits," and
LCO 3.4.2 "Minimum Temperature for Criticality,"

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 kept ≥ 541°F, and SDM is within the limits provided in the COLR.

PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(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.

Reference 7 allows special test exceptions (STE) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown 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, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 541°F,

b. SDM is within the limits provided in the COLR, and

c. THERMAL POWER is < 5% RTP.

BASES

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 enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

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.

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.

C.1

When the RCS lowest T_{avg} is < 541°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 541°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Action of Condition C cannot be completed within the associated Completion Time, the plant must be placed in a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, to reach MODE 3 from MODE 2 HZP conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1 "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is ≥ 541°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will provide assurance that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

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 3.1.8.4

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;

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 on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

2. 10 CFR 50.59, "Changes, Tests and Experiments."

3. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978.

4. ANSI/ANS-19.6.1-1997, "Reload Startup Physics Tests for Pressurized Water Reactors," American National Standards Institute, August 22, 1997.

5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

6. Chapter 14, "Initial Testing Program."

7. WCAP-11618, including Addendum 1, April 1989.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 Chemical and Volume Control System (CVS) Demineralized Water Isolation Valves and Makeup Line Isolation Valves

BASES

BACKGROUND	One of the principle functions of the CVS system is to maintain the reactor coolant chemistry conditions by controlling the concentration of boron in the coolant for plant startups, normal dilution to compensate for fuel depletion, and shutdown boration. In the dilute mode of operation, unborated demineralized water may be supplied directly to the reactor coolant system. Although the CVS is not considered a safety related system, certain functions of the system are considered safety related functions. The appropriate components have been classified and designed as safety related. The safety related functions provided by the CVS include containment isolation of chemical and volume control system lines penetrating containment, termination of inadvertent boron dilution, and preservation of the Reactor Coolant System (RCS) pressure boundary, including isolation of CVS letdown from the RCS.
APPLICABLE SAFETY ANALYSES	One of the initial assumptions in the analysis of an inadvertent boron dilution event (Ref. 1) is the assumption that the increase in core reactivity, created by the dilution event, can be detected by the source range instrumentation. The source range instrumentation will then supply a signal to the demineralized water isolation valves and the makeup line isolation valves in the CVS causing these valves to close and terminate the boron dilution event. Thus the makeup line isolation valves and the demineralized water isolation valves are components which function to mitigate or prevent an AOO. CVS isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The requirement that at least two demineralized water isolation valves and two makeup line isolation valves be OPERABLE assures that there will be redundant means available to terminate or prevent an inadvertent boron dilution event.
APPLICABILITY	The requirement that at least two demineralized water isolation valves and two makeup line isolation valves be OPERABLE is applicable in MODES 1, 2, 3, 4, and 5 because a boron dilution event is considered possible in these MODES, and the automatic closure of these valves is assumed in the safety analysis.

APPLICABILITY (continued)

In MODES 1 and 2, the detection and mitigation of a boron dilution event does not assume the detection of the event by the source range instrumentation. In these MODES, the event would be signalled by an intermediate range trip, a trip on the Power Range Neutron Flux - High (low setpoint nominally at 25% RTP), or Overtemperature delta T. The two demineralized water isolation valves close automatically upon reactor trip.

In MODE 6, a dilution event is precluded by the requirement in LCO 3.9.2 to close, lock and secure at least one valve in each unborated water source flow path.

ACTIONS

A.1

If only one of the demineralized water isolation valve and/or the makeup line isolation valve is/are OPERABLE, the redundant valve must be restored to OPERABLE status in 72 hours. The allowed Completion Time assures expeditious action will be taken, and is acceptable because the safety function of automatically isolating the clean water source can be accomplished by the redundant isolation valve(s).

B.1

If the Required Actions and associated Completion Time of Condition A are not met, or if both CVS demineralized water isolation valves or both makeup line isolation valves are not OPERABLE (i.e., not able to be closed automatically), then the demineralized water supply flow path to the RCS must be isolated. Isolation can be accomplished by manually isolating the CVS demineralized water isolation valve(s) or by positioning the 3-way blend valve to only take suction from the boric acid tank. Alternatively, the dilution path may be isolated by closing appropriate isolation valve(s) in the flow path(s) from the demineralized water storage tank to the reactor coolant system.

The Action is modified by a Note allowing the flow path 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 main control room. In this way, the flow path can be rapidly isolated when a need for isolation is indicated.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that the CVS demineralized water isolation valves and makeup line isolation valves are OPERABLE, by stroking each valve closed, demonstrates that the valves can perform their safety related function. The Frequency is in accordance with the Inservice Testing Program.

REFERENCES

1. Chapter 15, “Accident Analysis.”

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation with the On-line Power Distribution Monitoring System (OPDMS) inoperable, the global power distribution is limited by LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” and LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” which are directly and continuously measured process variables. These LCOs along with LCO 3.1.6, “Control Bank Insertion Limits,” maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

With the OPDMS OPERABLE, peak kw/ft (Z) (which is proportional to $F_Q(Z)$) is measured continuously. With the OPDMS inoperable, $F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

With the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$ with the OPDMS inoperable. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ which are present during a nonequilibrium situation such as load following.

To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor to account for the calculated worst case transient conditions.

Core monitoring and control under non-equilibrium conditions and the OPDMS inoperable are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed a limit of 2200°F (Ref. 1);

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 the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting (i.e., lower than) relative to the $F_Q(Z)$ assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} \qquad \text{for } P \leq 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

BASES

LCO (continued)

The actual values of CFQ are given in the COLR; however, CFQ is normally a number on the order of 2.60. For the AP1000, the normalized $F_Q(Z)$ as a function of core height is 1.0.

For RAOC operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results the measured value of $F_Q(Z)$, called $F_Q^M(Z)$ is obtained. Then,

$$F_Q^C(Z) = F_Q^M(Z) * F_Q^{MU}(Z)$$

where $F_Q^{MU}(Z)$ is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty. $F_Q^{MU}(Z)$ is provided in the COLR.

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

where W(Z) is a cycle-dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required and if $F_Q^W(Z)$ cannot be maintained within LCO limits, reduction of the AFD limits will also result in a reduction of the core power.

BASES

LCO (continued)

Violating the LCO limits for $F_Q(Z)$ may result in an unanalyzed condition while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

When the OPDMS is inoperable and core power distribution parameters cannot be continuously monitored, it is necessary to determine $F_Q(Z)$ on a periodic basis. Furthermore, the $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% of RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for fuel manufacturing tolerances and flux map measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

A reduction of the Power Range Neutron Flux – High Trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range

BASES

ACTIONS (continued)

Neutron Flux – High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux – High trip setpoint reductions within 8 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux – High trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux – High trip setpoints.

A.3

Reduction in the Overpower ΔT Trip setpoints (value of K_4) by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Overpower ΔT trip setpoints.

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, assures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Condition A is modified by a Note that requires Required Action A.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

BASES

ACTIONS (continued)

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ which can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours restricts the axial flux distribution such that even if a transient occurred, core peaking factors would not be exceeded.

The implicit assumption is that if W(Z) values were recalculated (consistent with the reduced AFD limits), then $F_Q^C(Z)$ times the recalculated W(Z) values would meet the $F_Q(Z)$ limit. Note that complying with this action (of reducing AFD limits) may also result in a power reduction. Hence the need for B.2, B.3, and B.4.

B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.

B.3

Reduction in the Overpower ΔT trip setpoints value of K_4 by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.

BASES

ACTIONS (continued)

B.4

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the maximum allowable power limit imposed by Required Action B.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Condition B is modified by a Note that requires Required Action B.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition A is exited prior to performing Required Action B.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

C.1

If Required Actions A.1 through A.4 or B.1 through B.4 are not met within their associated Completion Times, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by two Notes. The first note applies to the situation where the OPDMS is inoperable at the beginning of cycle startup. Note 1 applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% of RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are made at a lower power level at

BASES

SURVEILLANCE REQUIREMENTS (continued)

which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

The second Note applies to the situation where the OPDMS becomes inoperable while the plant is in MODE 1. Without the continuous monitoring capability of the OPDMS, F_Q limits must be monitored on a periodic basis. The first measurement must be made within 31 days of the most recent date where the OPDMS data has verified peak kw/ft (Z) (and therefore also F_Q) to be within its limit. This is consistent with the 31 day Surveillance Frequency.

<u>SR 3.2.1.1</u>

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing the measured values of $F_Q^C(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z) * F_Q^{MU}(Z)$. $F_Q^C(Z)$ is then compared to its specified limits.

The limit to which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP assures that the $F_Q^C(Z)$ limit is met when RTP is achieved because Peaking Factors generally decrease as power level is increased.

BASES

SURVEILLANCE REQUIREMENTS (continued)

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to assure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 effective full power days (EFPDs) is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with Technical Specifications.

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit to which $F_Q^W(Z)$ is compared varies inversely with power.

The W(Z) curve is provided in the COLR for discrete core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0% to 15% inclusive; and

b. Upper core region, from 85% to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the difficulty of making a precise measurement in these regions and because of the low probability that these regions would be more limiting than the safety analyses.

This Surveillance has been modified by a Note, which may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated and found to be within its limit, an evaluation of the expression below is

BASES

SURVEILLANCE REQUIREMENTS (continued)

required to account for any increase to $F_Q^M(Z)$ which could occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent FQ(Z) evaluations show an increase in $F_Q^C(Z)$, it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the greater of a factor of 1.02 or by an appropriate factor as specified in the COLR or to evaluate $F_Q(Z)$ more frequently, each 7 EFPDs. These alternative requirements will prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% of RTP ensures that the $F_Q(Z)$ limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

The Surveillance Frequency of 31 EFPDs is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with Technical Specifications, to preclude the occurrence of adverse peaking factors between 31 EFPD Surveillances. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

$F_Q(Z)$ is verified at power increases of at least 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions, to assure that $F_Q(Z)$ will be within its limit at higher power levels.

REFERENCES

1. 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors,” 1974.
2. Regulatory Guide 1.77, Rev. 0, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,” May 1974.
3. 10 CFR 50, Appendix A, GDC 26.
4. WCAP-7308-L-P-A, “Evaluation of Nuclear Hot Channel Factor Uncertainties,” June 1988 (Westinghouse Proprietary) and WCAP-7308-L-A (Non-Proprietary).
5. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification,” February 1994 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

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. 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 assures 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 integral of the linear power along the fuel rod with the highest integrated power 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.

With the On-line Power Distribution Monitoring System (OPDMS) OPERABLE, $F^N_{\Delta H}$ is determined continuously by the OPDMS. When the OPDMS is inoperable, $F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 effective full power days (EFPDs). Also, during power operation with the OPDMS inoperable, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio. Transient

BASES

BACKGROUND (continued)

events that may be DNB limited are assumed to begin with a $F^N_{\Delta H}$ that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ prevent core power distributions from occurring which would exceed the following fuel design limits:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a large break loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when the control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System (RCS) flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNB ratio (DNBR) to the 95/95 DNB criterion. This value provides a high degree of assurance that the hottest fuel rod in the core will not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this

BASES

APPLICABLE SAFETY ANALYSES (continued)

variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

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. 3).

The fuel is protected in part by Technical Specifications, which provide assurance that the initial conditions assumed in the safety and accident analyses remain valid. With the OPDMS OPERABLE, peak kw/ft(Z) and $F^N_{\Delta H}$ are directly monitored. Should the OPDMS become inoperable, the following LCOs assure that the conditions assumed for the safety analysis remain valid: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

When the OPDMS is not available to measure power distribution parameters continuously, $F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the incore detector system. Measurements are generally taken with the core at, or near, steady-state conditions. Without the OPDMS, core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided 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.

BASES

LCO (continued)

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.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

When the OPDMS is inoperable and core power distribution parameters cannot be continuously monitored, it is necessary to monitor $F^N_{\Delta H}(Z)$ on a periodic basis. Furthermore, $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and peak cladding temperature (PCT). Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power-dependent limit.

When the $F^N_{\Delta H}$ limit is exceeded, it is not likely that the DNBR limit would be violated in steady state operation, since events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain outside $F^N_{\Delta H}$ limits for an extended period of time.

BASES

ACTIONS (continued)

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 would nevertheless require another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

However, if power were reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 would be performed if power ascension were delayed past 24 hours.

<u>A.1.2.1 and A.1.2.2</u>

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux - High to ≤ 55% RTP in accordance with Required Action A.1.2.2. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those specified in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Time of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may cause an inadvertent reactor trip.

<u>A.2</u>

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because

BASES

ACTIONS (continued)

of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence assures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation will proceed within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% of RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note, that states that THERMAL POWER does not have to be reduced prior to performing this action.

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 8 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

When the OPDMS is OPERABLE, the value of $F^N_{\Delta H}$ is directly and continuously monitored. With the OPDMS inoperable, the value of $F^N_{\Delta H}$ is determined by using the incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by a measurement uncertainty factor before making comparisons to the $F^N_{\Delta H}$ limit.

After each refueling, with the OPDMS inoperable, $F^N_{\Delta H}$ must be determined prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

BASES

SURVEILLANCE REQUIREMENTS (continued)

With the OPDMS inoperable, the 31 EFPDs Frequency is acceptable because the power distribution will change relatively slowly over this amount of fuel burnup. This Frequency is short enough so that the $F^N_{\Delta H}$ limit will not be exceeded for any significant period of operation.

REFERENCES

1. Regulatory Guide 1.77, Rev. 0, May 1979.
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.”

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core when the On-Line Power Distribution Monitoring System (OPDMS) is inoperable. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing which is a significant factor in axial power distribution control.

RAOC is a calculational procedure which defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to assure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the computer which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

Although the RAOC defines limits that must be met to satisfy safety analyses, typically, without the OPDMS, an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day-to-day operation (Ref. 1). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup-dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

BASES

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing SAFETY to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

Three dimensional power distribution calculations are performed to demonstrate that normal operation power shapes are acceptable for the LOCA, the loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

With the OPDMS inoperable, the limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the LOCA. The most important Condition 3 event is the loss of flow accident. The most important Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System (CVS) to change boron concentration or from power level changes.

Signals are available to the operator from the Protection and Safety Monitoring System (PMS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

BASES

LCO (continued)

The AFD limits are provided in the COLR. Figure B 3.2.3-1 shows typical RAOC AFD limits. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

Violating this LCO on the AFD, with the OPDMS inoperable, could produce unacceptable consequences if a Condition 2, 3 or 4 event occurs while the AFD is outside its specified limits.

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP where the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES. With the OPDMS inoperable, it is necessary to monitor AFD via the excore detectors to ensure that it remains within the RAOC limits.

ACTIONS

A.1

Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition where the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This surveillance verifies that the AFD, as indicated by the PMS excore channel, is within its specified limits. The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. WCAP-8385, “Power Distribution Control and Load Following Procedures,” Westinghouse Electric Corporation, September 1974 (Westinghouse Proprietary) and WCAP-8403 (Non-Proprietary).

2. R.W. Miller et al., “Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification,” WCAP-10216-P-A, June 1983 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

3. Chapter 15, “Accident Analysis.”

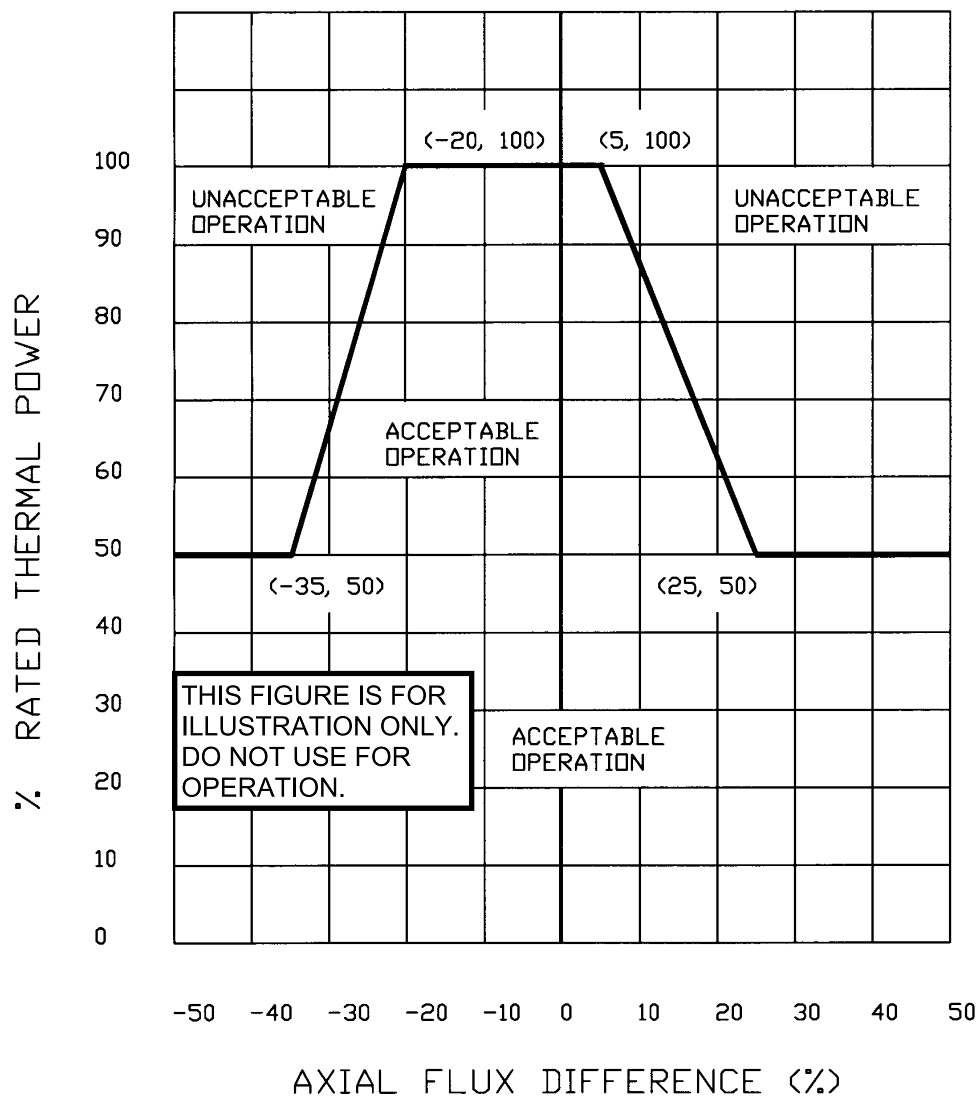

Figure B 3.2.3-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Limits as a Function
of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

With the OPDMS inoperable, the QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. With the OPDMS OPERABLE, the peak kw/ft(Z) is continuously and directly monitored. With the OPDMS inoperable, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions from occurring which would violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

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 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the 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.

APPLICABLE SAFETY ANALYSES (continued)

Should the OPDMS become inoperable, the QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, with the OPDMS inoperable, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, where corrective action is required, provides a margin of protection for both the DNB ratio (DNBR) and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to preclude core power distributions from exceeding the design limits. With the OPDMS inoperable, a continuous on-line indication of core peaking factors is not available. Therefore, QPTR must be monitored and the limits on QPTR ensure that peaking factors will be within design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, and the OPDMS inoperable, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

BASES

ACTIONS (continued)

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level and increasing power up to this revised limit.

A.2

After completion of Required Action A.1, the QPTR alarm may be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are of primary importance in assuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}^N$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction power Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction power Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limits, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}^N$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such

BASES

ACTIONS (continued)

reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors which best characterize the core power distribution. This re-evaluation is required to assure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR is not restored to within limits until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly,

BASES

ACTIONS (continued)

then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve the status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR as indicated by the Protection and Safety Monitoring System (PMS) excore channels is within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

For those causes of QPT that occur quickly (a dropped rod), there are other indications of abnormality that prompt a verification of core power tilt.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

With a PMS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts would likely be detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for assuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the incore detectors are used to confirm that the normalized symmetric power distribution is acceptable.

With the OPDMS and one PMS channel inoperable, the surveillance of the incore power distribution on a 12 hour basis is sufficient to maintain peaking factors within their normal limits, especially, considering the other LCOs and ACTIONS required when the OPDMS is out of service.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.”

2. Regulatory Guide 1.77, Rev. 0, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,” May 1974.

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” GDC 26, “Reactivity Control System Redundancy and Capability.”

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 OPDMS-Monitored Parameters

BASES

BACKGROUND

The On-line Power Distribution Monitoring System (OPDMS) for the AP1000 is an advanced core monitoring and support package. The OPDMS has the ability to continuously monitor core power distribution parameters. In addition, the OPDMS monitors SDM.

The purpose of the limits on the OPDMS-monitored power distribution parameters is to provide assurance of fuel integrity during Conditions I (Normal Operation) and II (incidents of Moderate Frequency) events by: (1) not exceeding the minimum departure from boiling ratio (DNBR) in the core, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the peak cladding temperature (PCT) limit of 2200°F is not exceeded.

The definition of certain quantities used in these specifications are as follows:

Peak kw/ft(Z)	Peak linear power density (axially dependent) as measured in kw/ft.
$F^N_{\Delta H}$	Ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.
Minimum DNBR	Minimum ratio of the critical heat flux to actual heat flux at any point in the reactor that is allowed in order to assure that certain performance and safety criteria requirements are met over the range of plant conditions.

By continuously monitoring the core and following its actual operation, it is possible to significantly limit the adverse nature of power distribution initial conditions for transients which may occur at any time.

APPLICABLE SAFETY ANALYSES

The limits on the above parameters preclude core power distributions from occurring which would violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the PCT must not exceed a limit of 2200°F (Ref. 1);

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 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

Limits on linear power density or peak kw/ft assure that the peak linear power density assumed as a base condition in the LOCA analyses is not exceeded during normal operation.

Limits on $F_{\Delta H}$ ensure that the LOCA analysis assumptions and assumptions made with respect to the Overtemperature ΔT Setpoint are maintained.

The limit on DNBR ensures that if transients analyzed in the safety analyses initiate from the conditions within the limit allowed by the OPDMS, the DNB criteria will be met.

The OPDMS-monitored power distribution parameters of this LCO satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within these limits. If the LCO limits cannot be maintained within limits, reduction of the core power is required.

Violating the OPDMS-monitored power distribution parameter limits could result in unanalyzed conditions should a design basis event occur while the parameters are outside their specified limits.

LCO (continued)

Peak kw/ft limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA. The highest calculated linear power densities in the core at specific core elevations are displayed for operator visual verification relative to the COLR values.

The determination of $F^N_{\Delta H}$ identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for DNB. Should $F^N_{\Delta H}$ exceed the limit given in the COLR, the possibility exists for DNBR to exceed the value used as a base condition for the safety analysis.

Two levels of alarms on power distribution parameters are provided to the operator. One serves as a warning before the three parameters (kw/ft(Z), $F^N_{\Delta H}$, DNBR) exceed their values used as a base condition for the safety analysis. The other alarm indicates when the parameters have reached their limits.

APPLICABILITY

The OPDMS-monitored power distribution parameter limits must be maintained in MODE 1 above 50% RTD to preclude core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES, and MODE 1 below 50% RTP, is not required because there is either insufficient stored energy in the fuel or insufficient energy transferred to the reactor coolant to require a limit on the distribution of core power. The OPDMS monitoring of SDM must be OPERABLE in MODES 1 and 2.

Specifically for $F^N_{\Delta H}$, the design bases accidents (DBAs) that are sensitive to $F^N_{\Delta H}$ in other MODES (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

In addition to the alarms discussed in the LCO section above (alarms on OPDMS-monitored power distribution parameters), there is an alarm indicating the potential inoperability of the OPDMS itself.

Should the OPDMS be determined to be inoperable for other than reasons of alarms inoperable, this LCO is no longer applicable and LCOs 3.2.1 through 3.2.4 become applicable.

ACTIONS

A.1

With any of the OPDMS-monitored power distribution parameters outside of their limits, the assumptions used as most limiting base conditions for the DBA analyses may no longer be valid. The 1 hour operator ACTION requirement to restore the parameter to within limits is consistent with the basis for the anticipated operational occurrences and provides time to assess if there are instrumentation problems. It also allows the possibility to restore the parameter to within limits by rod cluster control assembly (RCCA) motion if this is possible. The OPDMS will continuously monitor these parameters and provide an indication when they are approaching their limits.

B.1

If the OPDMS-monitored power distribution parameters cannot be restored to within their limits within the Completion Time of ACTION A.1, it is likely that the problem is not due to a failure of instrumentation. Most of these parameters can be brought within their respective limits by reducing THERMAL POWER because this will reduce the absolute power density at any location in the core thus providing margin to the limit.

If the parameters cannot be returned to within limits as power is being reduced, THERMAL POWER must be reduced to < 50% RTP where the LCOs are no longer applicable.

A Note has been added to indicate that if the power distribution parameters in violation are returned to within their limits during the power reduction, then power operation may continue at the power level where this occurs. This is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions.

The Completion Time of 4 hours provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain outside the $F_{\Delta H}^{N}$ limits for an extended period of time.

C.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

ACTIONS (continued)

as possible, the boron concentration should be a concentrated solution. The operator should begin boration with the best source available for the plant conditions.

SURVEILLANCE REQUIREMENTS

With OPDMS operating, the power distribution parameters are continuously computed and displayed, and compared against their limit. Two levels of alarms are provided to the operator. The first alarm provides a warning before these parameters (kw/ft(Z), $F^N_{\Delta H}$, and DNBR) exceed their limits. The second alarm indicates when they actually reach their limits. A third alarm indicates trouble with the OPDMS system.

SR 3.2.5.1

This Surveillance requires the operator to verify that the power distribution parameters are within their limits. This confirmation is a verification in addition to the automated checking performed by the OPDMS system. A 24 hour Surveillance interval provides assurance that the system is functioning properly and that the core limits are met.

With the OPDMS parameter alarms inoperable, an increased Surveillance Frequency is provided to assure that parameters are not approaching the limits. A 12 hour Frequency is adequate to identify changes in these parameters that could lead to their exceeding their limits.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors,” 1974.

2. Regulatory Guide 1.77, Rev. 0, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,” May 1974.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based upon the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Feature Actuation System (ESFAS) in mitigating accidents.

The Protection and Safety Monitoring System (PMS) has been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

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

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, “Safety Limits (SLs),” also maintains the above values and assures that offsite doses are within the acceptance criteria during AOOs.

Design Basis Accidents (DBA) are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of the limits. Different accident categories are allowed a different fraction of these limits, based on the probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RTS maintains surveillance on key process variables which are directly related to equipment mechanical limitations, such as pressure, and on variables which directly affect the heat transfer capability of the reactor, such as flow and temperature. Some limits, such as Overtemperature ΔT, are calculated in the protection and safety

BACKGROUND (continued)

monitoring system cabinets from other parameters when direct measurement of the variable is not possible.

The RTS instrumentation is segmented into four distinct but interconnected modules as identified below:

- Field inputs from process sensors, nuclear instrumentation;
- Protection and Safety Monitoring System Cabinets;
- Voting Logic; and
- Reactor Trip Switchgear Interface.

Field Transmitters and Sensors

Normally, four redundant measurements using four separate sensors are made for each variable used for reactor trip. The use of four channels for protection functions is based on a minimum of two channels being required for a trip or actuation, one channel in test or bypass, and a single failure on the remaining channel. The signal selector algorithm in the Plant Control System (PLS) will function with only three channels. This includes two channels properly functioning and one channel having a single failure. For protection channels providing data to the control system, the fourth channel permits one channel to be in test or bypass. Minimum requirements for protection and control is achieved with only three channels OPERABLE. The fourth channel is provided to increase plant availability, and permits the plant to run for an indefinite time with a single channel out of service. The circuit design is able to withstand both an input failure to the control system, which may then require the protection Function actuation, and a single failure in the other channels providing the protection Function actuation. Again, a single failure will neither cause nor prevent the protection Function actuation. These requirements are described in IEEE-603 (Ref. 5). The actual number of channels required for each plant parameter is specified in Reference 2.

Selected analog measurements are converted to digital form by digital converters within the protection and safety monitoring system cabinets. Signal conditioning may be applied to selected inputs following the conversion to digital form. Following necessary calculations and processing, the measurements are compared against the applicable setpoint for that variable. A partial trip signal for the given parameter is generated if one channel measurement exceeds its predetermined or calculation limit. Processing on all variables for reactor trip is duplicated in each of the four redundant divisions of the protection system. Each

BACKGROUND (continued)

division sends its partial trip status to each of the other three divisions over isolated multiplexed links. Each division is capable of generating a reactor trip signal if two or more of the redundant channels of a single variable are in the partial trip state.

The reactor trip signal from each division is sent to the corresponding reactor trip actuation division. Each of the four reactor trip actuation divisions consists of two reactor trip circuit breakers. The reactor is tripped when two or more actuation divisions receive a reactor trip signal. This automatic trip demand initiates the following two actions:

1. It de-energizes the undervoltage trip attachment on each reactor trip breaker, and

2. It energizes the shunt trip device on each reactor trip breaker.

Either action causes the breakers to trip. Opening of the appropriate trip breakers removes power to the control rod drive mechanism (CRDM) coils, allowing the rods to fall into the core. This rapid negative reactivity insertion shuts down the reactor.

<u>Protection and Safety Monitoring System Cabinets</u>

The protection and safety monitoring system cabinets contain the necessary equipment to:

- Permit acquisition and analysis of the sensor inputs, including plant process sensors and nuclear instrumentation, required for reactor trip and ESF calculations;

- Perform computation or logic operations on variables based on these inputs;

- Provide trip signals to the reactor trip switchgear and ESF actuation data to the ESF coincidence logic as required;

- Permit manual trip or bypass of each individual reactor trip Function and permit manual actuation or bypass of each individual voted ESF Function;

- Provide data to other systems in the Instrumentation and Control (I&C) architecture;

- Provide separate input circuitry for control Functions that require input from sensors that are also required for protection Functions.

BASES

BACKGROUND (continued)

Each of the four divisions provides signal conditioning, comparable output signals for indications in the main control room, and comparison of measured input signals with established setpoints. The basis of the setpoints are described in References 1, 2, and 3. If the measured value of a unit parameter exceeds the predetermined setpoint, an output is generated which is transmitted to the ESF coincidence logic for logic evaluation.

Within the protection and safety monitoring system redundancy is generally provided for active equipment such as processors and communication hardware. This redundancy is provided to increase plant availability and facilitate surveillance testing. A division or channel is OPERABLE if it is capable of performing its specified safety function(s) and all the required supporting functions or systems are also capable of performing their related support functions. Thus, a division or channel is OPERABLE as long as one set of redundant components within the division or channel is capable of performing its specified safety function(s).

Voting Logic

The voting logic provides a reliable means of opening the reactor trip switchgear in its own division as demanded by the individual protection functions.

Reactor Trip Switchgear Interface

The final stage of the voting logic provides the signal to energize the undervoltage trip attachment on each RTB within the reactor trip switchgear. Loss of the signal de-energizes the undervoltage trip attachments and results in the opening of those reactor trip switchgear. An additional external relay is de-energized with loss of the signal. The normally closed contacts of the relay energize the shunt trip attachments on each switchgear at the same time that the undervoltage trip attachment is de-energized. This diverse trip actuation is performed external to the PMS cabinets. The switchgear interface including the trip attachments and the external relay are within the scope of the PMS. Separate outputs are provided for each switchgear. Testing of the interface allows trip actuation of the breakers by either the undervoltage trip attachment or the shunt trip attachment.

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the trip output is set. Any trip output is considered to be properly adjusted when the “as left”

BASES

BACKGROUND (continued)

value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration accuracy).

The Trip Setpoints used in the trip output are based on the analytical limits stated in Reference 1. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), the Trip Setpoints and Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the “Westinghouse Setpoint Methodology for Protection Systems” (Refs. 4 and 9). The actual nominal Trip Setpoint entered into the trip output is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the trip output is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed). Note that in the accompanying LCO 3.3.1, the Trip Setpoints of Table 3.3.1-1 are the LSSS.

The Trip Setpoints and Allowable Values listed in Table 3.3.1-1 are based on the methodology described in Reference 4, which incorporates all of the known uncertainties applicable for each channel. (Reference 4 is an AP600 document that describes a methodology that is applicable to AP1000. AP1000 has some slight differences in instrument spans as a result of the higher power level.) The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes. Transmitter and signal processing equipment calibration tolerances and drift allowances must be specified in plant calibration procedures, and must be consistent with the values used in the setpoint methodology.

The OPERABILITY of each transmitter or sensor can be evaluated when its “as found” calibration data are compared against the “as left” data and are shown to be within the setpoint methodology assumptions. The basis

BACKGROUND (continued)

of the setpoints is described in References 1, 2, 3, and 4. Trending of transmitter calibration is required by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle."

The protection and safety monitoring system testing features are designed to allow for complete functional testing by using a combination of system self-checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded. For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing. To the extent possible, protection and safety monitoring system functional testing will be accomplished with continuous system self-checking features and the continuous functional testing features.

The protection and safety monitoring system incorporates continuous system self-checking features wherever practical. Self-checking features include on-line diagnostics for the computer system and the hardware and communications tests. These self-checking tests do not interfere with normal system operation.

In addition to the self-checking features, the system includes functional testing features. Functional testing features include continuous functional testing features and manually initiated functional testing features. To the extent practical functional testing features are designed not to interfere with normal system operation.

In addition to the system self-checking features and functional testing features, other test features are included for those parts of the system which are not tested with self-checking features or functional testing features. These test features allow for instruments/sensor checks, calibration verification, response time testing, setpoint verification and component testing. The test features again include a combination of continuous testing features and manual testing features.

All of the testing features are designed so that the duration of the testing is as short as possible. Testing features are designed so that the actual logic is not modified. To prevent unwanted actuation, the testing features are designed with either the capability to bypass a Function during testing and/or limit the number of signals allowed to be placed in test at one time.

BACKGROUND (continued)

Reactor Trip (RT) Channel

An RT Channel extends from the sensor to the output of the associated reactor trip subsystem in the protection and safety monitoring system cabinets, and includes the sensor (or sensors), the signal conditioning, any associated datalinks, and the associated reactor trip subsystem. For RT Channels containing nuclear instrumentation, the RT Channel also includes the nuclear instrument signal conditioning and the associated Nuclear Instrumentation Signal Processing and Control (NISPAC) subsystem.

Automatic Trip Logic

The Automatic Trip Logic extends from, but does not include, the outputs of the various RT Channels to, but does not include, the reactor trip breakers. Operator bypass of a reactor trip function is performed within the Automatic Trip Logic.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The RTS functions to maintain the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

Each of the analyzed accidents and transients which require reactor trip can be detected by one of more RTS Functions. The accident analysis described in Reference 3 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These RTS trip Functions may provide protection for conditions which do not require dynamic transient analysis to demonstrate function performance. These RTS trip Functions may also serve as backups to RTS trip Functions that were credited in the accident analysis.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of three channels in each instrumentation Function.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the main control room operator can initiate a reactor trip at any time by using either of two reactor trip actuation devices in the main control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It can be used by the reactor operator to shutdown the reactor whenever any parameter is rapidly trending toward its Trip Setpoint. The safety analyses do not take credit for the Manual Reactor Trip.

The LCO requires two Manual Reactor Trip actuation devices be OPERABLE in MODES 1 and 2 and in MODES 3, 4, and 5 with RTBs closed and PLS capable of rod withdrawal. Two independent actuation devices are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if the shutdown or control rods are withdrawn or the PLS is capable of withdrawing the shutdown or control rods. In MODES 3, 4, and 5, manual initiation of a reactor trip does not have to be OPERABLE if the PLS is not capable of withdrawing the shutdown or control rods. If the rods cannot be withdrawn from the core, there is no need to be able to trip the reactor because all of the rods are inserted. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function does not have to be OPERABLE.

2. Power Range Neutron Flux

The PMS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The PMS power range detectors provide input to the PLS. Minimum requirements for protection and control is achieved with three channels OPERABLE. The fourth channel is provided to increase plant availability, and permits the plant to run for an

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

indefinite time with a single channel in trip or bypass. This Function also satisfies the requirements of IEEE 603 (Ref. 5) with 2/4 logic. This Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

a. Power Range Neutron Flux – High

The Power Range Neutron Flux – High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion during power operations. Positive reactivity excursions can be caused by rod withdrawal or reductions in RCS temperature.

The LCO requires four Power Range Neutron Flux – High channels to be OPERABLE in MODES 1 and 2.

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux – High trip must be OPERABLE. This Function will terminate the reactivity excursion and shutdown the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux – High trip does not have to be OPERABLE because the reactor is shutdown and a reactivity excursion in the power range cannot occur. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6. In addition, the PMS power range detectors cannot detect neutron levels in this range.

b. Power Range Neutron Flux – Low

The LCO requirement for the Power Range Neutron Flux – Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions. The Trip Setpoint reflects only steady state instrument uncertainties as this Function does not provide primary protection for any event that results in a harsh environment.

The LCO requires four of the Power Range Neutron Flux – Low channels to be OPERABLE in MODE 1 below the Power Range Neutron Flux P-10 Setpoint and MODE 2.

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In MODE 1, below the Power Range Neutron Flux P-10 setpoint and in MODE 2, the Power Range Neutron Flux – Low trip must be OPERABLE. This Function may be manually blocked by the operator when the respective power range channel is greater than approximately 10% of RTP (P-10 setpoint). This Function is automatically unblocked when the respective power range channel is below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux – High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux – Low trip Function does not have to be OPERABLE because the reactor is shutdown and the PMS power range detectors cannot detect neutron levels generated in MODES 3, 4, 5, and 6. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Power Range Neutron Flux – High Positive Rate

The Power Range Neutron Flux – High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux which are characteristic of a rod cluster control assembly (RCCA) drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux – High and Low trip Functions to ensure that the criteria are met for a rod ejection from the power range. The Power Range Neutron Flux Rate trip uses the same channels as discussed for Function 2 above.

The LCO requires four Power Range Neutron Flux – High Positive Rate channels to be OPERABLE. In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux – High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux – High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure

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bolts are detensioned preventing any pressure buildup. In addition, the PMS power range detectors cannot detect neutron levels present in this MODE.

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux – Low Setpoint trip Function. The PMS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The safety analyses do not take credit for the Intermediate Range Neutron Flux trip Function. Even though the safety analyses take no credit for the Intermediate Range Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of the RTS. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment. This trip can be manually blocked by the main control room operator when above the P-10 setpoint, which is the respective PMS power range channel greater than 10% power, and is automatically unblocked when below the P-10 setpoint, which is the respective PMS power range channel less than 10% power. This Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires four channels of Intermediate Range Neutron Flux to be OPERABLE. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 below the P-10 setpoint, and in MODE 2, when there is a potential for an uncontrolled rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux – High Setpoint trip and the Power Range Neutron Flux – High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the

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required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the PMS intermediate range detectors cannot detect neutron levels present in this MODE.

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux – Low Setpoint and Intermediate Range Neutron Flux trip Functions. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The PMS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The safety analyses do not take credit for the Source Range Neutron Flux trip Function. Even though the safety analyses take no credit for the Source Range Neutron Flux trip, the functional capability at the specified Trip Setpoint is assumed to be available and the trip is implicitly assumed in the safety analyses.

The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment. This trip can be manually blocked by the main control room operator when above the P-6 setpoint (Intermediate Range Neutron Flux interlock) and is automatically unblocked when below the P-6 setpoint. The manual block of the trip function also de-energizes the source range detectors. The source range detectors are automatically re-energized when below the P-6 setpoint. The trip is automatically blocked when above the P-10 setpoint (Power Range Neutron Flux interlock). The source range trip is the only RTS automatic protective Function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The LCO requires four channels of Source Range Neutron Flux to be OPERABLE in MODE 2 below P-6 and in MODE 3, 4, or 5 with RTBs closed and Control Rod Drive System capable of rod withdrawal. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODE 3, 4, or 5 with the RTBs open, the LCO does not require the Source Range Neutron Flux channels for reactor trip Functions to be OPERABLE.

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In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux – Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the PMS source range detectors are de-energized and inoperable as described above.

In MODE 3, 4, or 5 with the reactor shutdown, the Source Range Neutron Flux trip Function must also be OPERABLE. If the PLS is capable of rod withdrawal, the Source Range Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the PLS is not capable of rod withdrawal, the source range detectors are required to be OPERABLE to provide monitoring of neutron levels and provide protection for events like an inadvertent boron dilution. These Functions are addressed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." The requirements for the PMS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation."

6. Overtemperature ΔT

The Overtemperature ΔT trip Function ensures that protection is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include all combinations of pressure, power, coolant temperature, and axial power distribution, assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Overtemperature ΔT trip Function uses the measured T_{HOT} and T_{COLD} in each loop, together with the measured pressurizer pressure, to compute the reactor core thermal power. Equations to fit the properties of density and enthalpy are programmed in the software, such that the ΔT power signal is presented as a percent of RTP for direct comparison with measured calorimetric power. The overtemperature ΔT setpoint is automatically varied for changes in the parameters that affect DNB as follows:

- reactor core inlet temperature – the Trip Setpoint is varied to correct for changes in core inlet temperature based on measured changes in cold leg temperature with dynamic compensation to account for cold leg-to-core transit time;

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- pressurizer pressure – the Trip Setpoint is varied to correct for changes in system pressure; and

- axial power distribution – the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the PMS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower PMS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation of the ΔT power signal is included for system piping delays from the core to the temperature measurement system. The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. This Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip. No credit is taken in the safety analyses for the turbine runback.

The LCO requires four channels of the Overtemperature ΔT trip Function to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. <u>Overpower ΔT</u>

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip function and provides a backup to the Power Range Neutron Flux – High Setpoint trip. The Overpower ΔT

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trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the same ΔT power signal generated for the Overtemperature ΔT. The setpoint is automatically varied with the following parameter:

- Axial power distribution – the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the PMS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower PMS power range detectors, the Trip Setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. The Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties as the detectors provide protection for a steam line break and may be in a harsh environment. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback reduces turbine power and reactor power. A reduction in power normally alleviates the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The Overpower ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to a affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

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8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure – High and – Low trips and the Overtemperature ΔT trip.

a. Pressurizer Pressure – Low

The Pressurizer Pressure – Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure. The Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide primary protection for an event that results in a harsh environment.

The LCO requires four channels of Pressurizer Pressure – Low to be OPERABLE in MODE 1 above P-10. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure – Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-10 interlock. On decreasing power, this trip Function is automatically blocked below P-10. Below the P-10 setpoint, no conceivable power distributions can occur that would cause DNB concerns.

b. Pressurizer Pressure – High

The Pressurizer Pressure – High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the safety valves to prevent RCS overpressure conditions. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

The LCO requires four channels of the Pressurizer Pressure – High to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

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In MODE 1 or 2, the Pressurizer Pressure – High trip must be OPERABLE to help prevent RCS overpressurization and LCOs, and minimizes challenges to the safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure – High trip Function does not have to be OPERABLE because transients which could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate plant conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level – High 3

The Pressurizer Water Level – High 3 trip Function provides a backup signal for the Pressurizer Pressure – High 3 trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment. The level channels do not actuate the safety valves.

The LCO requires four channels of Pressurizer Water Level – High 3 to be OPERABLE in MODE 1 above P-10. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 when there is a potential for overfilling the pressurizer, the Pressurizer Water Level – High 3 trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-10 interlock. On decreasing power, this trip Function is automatically blocked below P-10. Below the P-10 setpoint, transients which could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate plant conditions and take corrective actions.

10. Reactor Coolant Flow – Low

a. Reactor Coolant Flow

The Reactor Coolant Flow – Low trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS hot legs. Above the P-10 setpoint, a loss of flow in any RCS hot leg will actuate a

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Reactor trip. Each RCS hot leg has four flow detectors to monitor flow. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

The LCO requires four Reactor Coolant Flow – Low channels per hot leg to be OPERABLE in MODE 1 above P-10. Four OPERABLE channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 above the P-10 setpoint, when a loss of flow in one RCS hot leg could result in DNB conditions in the core, the Reactor Coolant Flow – Low trip must be OPERABLE.

11. <u>Reactor Coolant Pump (RCP) Bearing Water Temperature – High</u>

    a. <u>RCP Bearing Water Temperature – High</u>

    The RCP Bearing Water Temperature – High reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS cold leg. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

    The LCO requires four RCP Bearing Water Temperature – High channels per RCP to be OPERABLE in MODE 1 or 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

    In MODE 1 or 2, when a loss of flow in any RCS cold leg could result in DNB conditions in the core, the RCP Bearing Water Temperature – High trip must be OPERABLE.

12. <u>Reactor Coolant Pump Speed – Low</u>

    The RCP Speed – Low trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS cold legs. The speed of each RCP is monitored. Above the P-10 setpoint a low speed detected on two or more RCPs

    will initiate a reactor trip. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

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The LCO requires four RCP Speed – Low channels to be OPERABLE in MODE 1 above P-10. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 above the P-10 setpoint, the RCP Speed – Low trip must be OPERABLE. Below the P-10 setpoint, all reactor trips on loss of flow are automatically blocked since no power distributions are expected to occur that would cause a DNB concern at this low power level. Above the P-10 setpoint, the reactor trip on loss of flow in two or more RCS cold legs is automatically enabled.

13. Steam Generator Water Level – Low

The SG Water Level – Low trip Function ensures that protection is provided against a loss of heat sink. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low level in any steam generator is indicative of a loss of heat sink for the reactor. The Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties as the detectors provide primary protection for an event that results in a harsh environment. This Function also contributes to the coincidence logic for the ESFAS Function of opening the Passive Residual Heat Removal (PRHR) discharge valves.

The LCO requires four channels of SG Water Level – Low per SG to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level – Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater System (non-safety related). The Main Feedwater System is normally in operation in MODES 1 and 2. PRHR is the safety related backup heat sink for the reactor. During normal startups and shutdowns, the Main and Startup Feedwater Systems (non-safety related) can provide feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level – Low Function does not have to be OPERABLE because the reactor is not operating or even critical.

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14. <u>Steam Generator Water Level – High 2</u>

    The SG Water Level – High 2 trip Function ensures that protection is provided against excessive feedwater flow by closing the main feedwater control valves, tripping the turbine, and tripping the reactor. While the transmitters (d/p cells) are located inside containment, the events which this function protects against cannot cause severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

    The LCO requires four channels of SG Water Level – High 2 per SG to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

    In MODES 1 and 2 above the P-11 interlock, the SG Water Level – High 2 trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater System (non-safety related). The Main Feedwater System is only in operation in MODES 1 and 2. In MODE 3, 4, 5, or 6, the SG Water Level – High 2 Function does not

    have to be OPERABLE because the reactor is not operating or even critical. The P-11 interlock is provided on this Function to permit bypass of the trip Function when the pressure is below P-11. This bypass is necessary to permit rod testing when the steam generators are in wet layup.

15. <u>Safeguards Actuation Signal from Engineered Safety Feature Actuation System</u>

    The Safeguards Actuation Signal from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal which initiates the Safeguards Actuation signal. This is a condition of acceptability for the Loss of Coolant Accident (LOCA). However, other transients and accidents take credit for varying levels of ESFAS performance and rely upon rod insertion, except for the most reactive rod which is assumed to be fully withdrawn, to ensure reactor shutdown.

    The LCO requires two manual and four automatic divisions of Safeguards Actuation Signal Input from ESFAS to be OPERABLE in MODES 1 and 2. Four automatic divisions are provided to permit one division bypass indefinitely and still ensure no single random failure will disable this trip Function.

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A reactor trip is initiated every time a Safeguards Actuation signal is present. Therefore, this trip Function must be OPERABLE in MODES 1 and 2, when the reactor is critical, and must be shutdown in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical.

16. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current plant status. They back up operator actions to ensure protection system Functions are not blocked during plant conditions under which the safety analysis assumes the Functions are OPERABLE. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip Functions are outside the applicable MODES.
These are:

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when the respective PMS Intermediate Range Neutron Flux channel goes approximately one decade above the minimum channel reading. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-6 interlock allows the manual block of the respective PMS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed.

(2) on decreasing power, the P-6 interlock automatically energizes the PMS source range detectors and enables the PMS Source Range Neutron Flux reactor trip.

(3) on increasing power, the P-6 interlock provides a backup block signal to the source range neutron flux doubling circuit. Normally, this Function is manually blocked by the main control room operator during the reactor startup.

The LCO requires four channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint.

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In MODE 2, when below the P-6 interlock setpoint, the P-6 interlock must be OPERABLE. Above the P-6 interlock setpoint, the PMS Source Range Neutron Flux reactor trip will be blocked; and this Function will no longer be necessary. In MODES 3, 4, 5, and 6, the P-6 interlock does not have to be OPERABLE because the PMS Source Range is providing core protection.

b. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power as determined by the respective PMS power-range detector. The LCO requirement for the P-10 interlock ensures that the following functions are performed:

(1) on increasing power, the P-10 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure – Low,
- Pressurizer Water Level – High 3,
- Reactor Coolant Flow – Low, and
- RCP Speed – Low.

These reactor trips are only required when operating above the P-10 setpoint (approximately 10% power). These reactor trips provide protection against violating the DNBR limit. Below the P-10 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip.

(3) on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux – Low Setpoint reactor trip.

(4) on increasing power, the P-10 interlock automatically provides a backup block signal to the Source Range Neutron Flux reactor trip and also to de-energize the PMS source range detectors.

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(5) on decreasing power, the P-10 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure – Low,
- Pressurizer Water Level – High 3,
- Reactor Coolant Flow – Low, and
- RCP Speed – Low.

(6) on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux – Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2.

In MODE 1, when the reactor is at power, the Power Range Neutron Flux, P-10 interlock must be OPERABLE. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux – Low Setpoint and Intermediate

Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

c. Pressurizer Pressure, P-11

With pressurizer pressure channels less than the P-11 setpoint, the operator can manually block the Steam Generator Narrow Range Water Level – High 2 reactor Trip. This allows rod testing with the steam generators in cold wet layup. With pressurizer pressure channels > P-11 setpoint, the Steam

Generator Narrow Range Water Level – High 2 reactor Trip is automatically enabled. The operator can also enable these actuations by use of the respective manual reset.

17. Reactor Trip Breakers

This trip Function applies to the RTBs exclusive of individual trip mechanisms. There are eight reactor trip breakers with two breakers in each division. The reactor trip circuit breakers are arranged in a

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two-out-of-four logic configuration, such that the tripping of the two circuit breakers associated with one division does not cause a reactor trip. This circuit breaker arrangement is illustrated in Figure 7.1-7. The LCO requires four divisions of the Reactor Trip Switchgear to be OPERABLE with two trip breakers associated with each required division. This logic is required to meet the safety function assuming a single failure.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs are closed, and the PLS is capable of rod withdrawal.

18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the PLS, or declared inoperable under Function 17 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening the breakers on a valid signal.

These trip Functions must be OPERABLE in MODES 1 and 2 when the reactor is critical. In MODES 3, 4, and 5, these RTS trip Functions must be OPERABLE when the RTBs are closed, and the PLS is capable of rod withdrawal.

19. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 17 and 18) and Automatic Trip Logic (Function 19) ensures that means are provided to interrupt the power to the CRDMs and allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed.

The automatic trip logic includes the ESF coincidence logic and the voting logic.

The LCO requires four divisions of RTS Automatic Trip Logic to be OPERABLE. Four OPERABLE divisions are provided to ensure that a random failure of a single logic channel will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs are closed and the PLS is capable of

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

rod withdrawal.

20. ADS Stages 1, 2 and 3 Actuation Input from Engineered Safety Feature Actuation System

    The LCO requirement for this Function provides a reactor trip for any event that may initiate depressurization of the reactor.

    The LCO requires four divisions of RTS Automatic Trip Logic to be OPERABLE. Four OPERABLE divisions are provided to ensure that a random failure of a single logic channel will not prevent reactor trip.

    These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs are closed and the PLS is capable of rod withdrawal.

21. Core Makeup Tank (CMT) Actuation Input from Engineered Safety Feature Actuation System

    The LCO requirement for this Function provides a reactor trip for any event that may initiate CMT injection.

    The LCO requires four divisions of RTS Automatic Trip Logic to be OPERABLE. Four OPERABLE divisions are provided to ensure that random failure of a single logic channel will not prevent reactor trip.

    These trip Functions must be OPERABLE in MODES 1 and 2 when the reactor is critical. In MODE 3, 4, and 5 these RTS trip Functions must be OPERABLE when the RTBs are closed and the PLS is capable of rod withdrawal.

The RTS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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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 on Table 3.3.1-1.

In the event the transmitter, instrument loop, signal processing electronics, or trip output is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO

Condition(s) entered for the protection Function(s) affected.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all RTS protection Functions. Condition A addresses the situation where one or more required channels for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection Functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1, B.2.1, and B.2.2

Condition B applies to the Manual Reactor Trip, Manual Safeguards Actuation, Manual ADS Stages 1, 2, and 3 Actuation and Manual Core Makeup Tank Actuation in MODES 1 and 2 and in MODES 3, 4, and 5 with the reactor trip breakers closed and the plant control system capable of rod withdrawal. These Required Actions address inoperability of one manual initiation device of the Manual Reactor Trip Function, Manual Safeguards Actuation Function, Manual ADS Stages 1, 2, and 3 Actuation Function and/or Manual Core Makeup Tank Actuation Function. One device consists of an actuation switch and the associated hardware (such as contacts and wiring) up to but not including the eight Reactor Trip Breakers. With one device inoperable, the inoperable device must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE device is adequate to perform the safety function.

If the manual Function(s) cannot be restored to OPERABLE status in the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours (54 hours

ACTIONS (continued)

total time) followed by opening the RTBs within 1 additional hour (55 hours total time). The 6 additional hours to reach MODE 3 and the 1 hour to open the RTBs are reasonable, based on operating experience, to reach MODE 3 and open the RTBs from full power operation in an orderly manner and without challenging unit systems. With the RTBs open and the unit in MODE 3, this trip Function is no longer required to be

ACTIONS (continued)

OPERABLE.

C.1 and C.2

Condition C applies to the Manual Reactor Trip in MODES 3, 4, and 5 with the RTBs closed and the PLS capable of rod withdrawal. These Required Actions address inoperability of one manual initiation device of the Manual Reactor Trip Function. One device consists of an actuation switch and the associated hardware (such as contacts and wiring) up to but not including the eight Reactor Trip Breakers. With one device inoperable, the inoperable device must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE device is adequate to perform the safety function.

If the Manual Reactor Trip Function cannot be restored to OPERABLE status in the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs must be opened within the next 1 hour. With the RTBs open, this Function is no longer required.

D.1.1, D.1.2, D.1.3, D.2.1, D.2.2, and D.3

Condition D applies to the Power Range Neutron Flux – High Function in MODES 1 and 2.

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within 6 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 7.

In addition to placing the inoperable channel(s) in the bypassed or tripped condition, THERMAL POWER must be reduced to ≤ 75% RTP within 12 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. With one or two of the PMS power range detectors inoperable, partial radial power distribution monitoring capability is lost. However, the protective function would still function even with a single failure of one of the two remaining channels.

As an alternative to reducing power, the inoperable channel(s) can be placed in the bypassed or tripped condition within 6 hours and the QPTR

ACTIONS (continued)

monitored every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR compensates for the lost monitoring capability and allows continued plant operation at power levels > 75% RTP. The 12 hour Frequency is consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

Required Action D.2.2 has been modified by a Note which only requires SR 3.2.4.2 to be performed if OPDMS and the Power Range Neutron Flux input to QPTR become inoperable. Power distribution limits are normally verified in accordance with LCO 3.2.5, "OPDMS - Monitored Power Distribution Parameters." However, if OPDMS becomes inoperable, then LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," becomes applicable. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. If either OPDMS or the channel input to QPTR is OPERABLE, then performance of SR 3.2.4.2 once per 12 hours is not necessary.

As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE.
Twelve hours are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

<u>E.1.1, E.1.2, and E.2</u>

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux – Low;
- Overtemperature ΔT;
- Overpower ΔT;
- Power Range Neutron Flux – High Positive Rate;
- Pressurizer Pressure – High;
- RCP Bearing Water Temperature – High;
- SG Water Level – Low; and
- SG Water Level – High 2.

With one or two channels inoperable, one affected channel must be

ACTIONS (continued)

placed in a bypass or trip condition within 6 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 7.

If the Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

F.1.1, F.1.2, F.2, and F.3

Condition F applies to the Intermediate Range Neutron Flux trip when above the P-6 setpoint and below the P-10 setpoint. Above the P-6 setpoint and below the P-10 setpoint, the PMS intermediate range detector performs the monitoring functions.

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within 2 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 2 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 7.

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As an alternative to placing the channel(s) in bypass or trip if THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 2 hours are allowed to reduce THERMAL POWER below the P-6 setpoint or to increase the THERMAL POWER above the P-10 setpoint. The PMS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the PMS power range detectors perform the monitoring and protective functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment below P-6, and takes into account the redundant capability afforded by the two remaining OPERABLE channels and the low probability of their failure during this period.

G.1 and G.2

Condition G applies to three Intermediate Range Neutron Flux trip channels inoperable in MODE 2 above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the PMS intermediate range detector performs the monitoring Functions. With only one intermediate range channel OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are insufficient

OPERABLE Intermediate Range Neutron Flux channels to adequately monitor the power rise. The operator must also reduce THERMAL POWER below the P-6 setpoint within 2 hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the PMS Intermediate Range Neutron Flux trip.

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H.1

Condition H applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is below the P-6 setpoint and one or two channels is inoperable. Below the P-6 setpoint, the PMS source range performs the monitoring and protective functions. At least three of the four PMS intermediate range channels must be returned to OPERABLE status prior to increasing power above the P-6 setpoint. With the unit in this Condition, below P-6, the PMS source range performs the monitoring and protection functions.

I.1

Condition I applies to one or two Source Range Neutron Flux trip channels inoperable when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the PMS source range performs the monitoring and protection functions. With one or two of the four channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only two source range channels OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

J.1

Condition J applies to three inoperable Source Range Neutron Flux channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With three source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition and the unit enters Condition T.

K.1.1, K.1.2, and K.2

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure – Low;
- Pressurizer Water Level – High 3;
- Reactor Coolant Flow – Low; and

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- RCP Speed – Low.

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within 6 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 7.

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. A Completion Time of an additional 6 hours is allowed to reduce power < P-10. Allowance of this time interval takes into consideration the redundant capability provided by the remaining two redundant OPERABLE channels and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

<u>L.1 and L.2</u>

Condition L applies to the Safeguards Actuation signal from ESFAS reactor trip, the RTS Automatic Trip Logic, automatic ADS Stages 1, 2, and 3 actuation, and automatic CMT injection in MODES 1 and 2.

With one or two channels or divisions inoperable, the Required Action is to restore three of the four channels/divisions within 6 hours. Restoring all channels/divisions but one to OPERABLE status ensures that a single failure will neither cause nor prevent the protective function. The 6 hour Completion Time is considered reasonable since the protective function will still function.

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. A Completion Time of an additional 6 hours is allowed to place the unit in MODE 3. The Completion Time is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Allowance of this time interval takes into consideration the redundant capability provided by the remaining two redundant OPERABLE channels/divisions and the low probability of occurrence of an event during this period that may require the protection

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afforded by this Function.

<u>M.1, M.2.1, M.2.2, and M.3</u>

Condition M applies to the P-6, P-10, and P-11 interlocks. With one or two channels inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour, or the Functions associated with inoperable interlocks placed in a bypassed or tripped condition within 7 hours, or the unit must be placed in MODE 3 within 13 hours. Verifying the interlock manually accomplishes the interlock condition.

If one interlock channel is inoperable, the associated Function(s) must be placed in a bypass or trip condition within 7 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.)

If two interlock channels are inoperable, one channel of the associated Function(s) must be bypassed and one channel of the associated Function(s) must be tripped. In this state, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 7 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 7.

If placing the associated Functions in bypass or trip is impractical, for instance as the result of other channels in bypass or trip, the Completion Time of an additional 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

<u>N.1, N.2.1, and N.2.2</u>

Condition N applies to the RTBs, and RTB undervoltage and shunt trip mechanisms in MODES 1 and 2, and in MODES 3, 4, and 5 with the RTBs closed and the PLS capable of rod withdrawal. This Condition is primarily associated with mechanical damage that can prevent the RTBs from opening.

With one division inoperable, the reactor trip breakers in the inoperable division must be opened within 8 hours. A division is inoperable, if, within that division, one or both of the RTBs and/or one or both of the trip mechanisms is inoperable.

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With one division inoperable (with its RTBs open) and with three OPERABLE divisions remaining, the trip logic becomes one-out-of-three. The one-out-of-three trip logic meets the single failure criterion. (A failure in one of the three remaining divisions will not prevent the protective function.) If, coincident with RTBs inoperable in one division, the automatic trip logic is inoperable in another division, the trip logic becomes one-out-of-two, which meets the single failure criterion.

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE within an additional 6 hours. This is done by opening all of the RTBs. With the RTBs open, these Functions are no longer required.

O.1, O.2.1, and O.2.2

Condition O applies to the RTBs in MODES 1 and 2, and in MODES 3, 4, and 5 with the RTBs closed and the PLS capable of rod withdrawal. With two divisions of RTBs and/or RTB Undervoltage and Shunt Trip Mechanisms inoperable, 1 hour is allowed to restore the three of the four divisions to OPERABLE status or the unit must be placed in MODE 3, 4 or 5 and the RTBs opened within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1-hour and 6-hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

P.1 and P.2

Condition P applies to automatic ADS Stages 1, 2, and 3 Actuation, automatic CMT Actuation and the RTS Automatic Trip Logic in MODES 3, 4, and 5 with the RTBs closed and the PLS capable of rod withdrawal.

With one or two channels/divisions inoperable, three of the four channels/divisions must be restored to OPERABLE status in 48 hours. Restoring all channels but one to OPERABLE ensures that a single failure will neither cause nor prevent the protective function. The 48 hour Completion Time is considered reasonable since the protective function will still function.

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. A Completion Time of an additional 1 hour is allowed to open the RTBs. With RTBs open, these

ACTIONS (continued)

Functions are no longer required.

Q.1 and Q.2

Condition Q applies to one or two inoperable Source Range Neutron Flux channels in MODE 3, 4, or 5 with the RTBs closed and the PLS capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one or two of the source range channels inoperable, 48 hours is allowed to restore three of the four channels to an OPERABLE status. If the channels cannot be returned to an OPERABLE status, 1 additional hour is allowed to open the RTBs. Once the RTBs are open, the core is in a more stable condition and the unit enters Condition R. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour to open the RTBs, are justified in Reference 7.

R.1, R.2, and R.3

Condition R applies when the required Source Range Neutron Flux channel is inoperable in MODE 3, 4, or 5 with the RTBs open. With the unit in this Condition, the NIS source range performs the monitoring and protection functions. With the required source range channel inoperable, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation. In addition to suspension of positive reactivity additions, all valves that could add unborated water to the RCS must be closed within 1 hour as specified in LCO 3.9.2. The isolation of unborated water sources will preclude a boron dilution accident.

Also, the SDM must be verified within 1 hour and once every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Required Action R.3 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Times of within 1 hour and once per 12 hours are based on operating experience in performing the Required Actions and the knowledge that unit conditions will change slowly.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified in the SRs column of Table 3.3.1-1 for that Function.

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ACTIONS (continued)

A Note has been added to the SR table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

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SURVEILLANCE REQUIREMENTS (continued)

The CHANNEL CALIBRATION and RTCOT are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies. For channels that include dynamic transfer functions, such as, lag, lead/lag, rate/lag, the response time test may be performed with the transfer function set to one, with the resulting measured response time compared to the appropriate Chapter 7 response time (Ref. 2). Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

<u>SR 3.3.1.1</u>

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment have drifted outside its limit.

The channels to be checked are:

Power Range Neutron Flux
Intermediate Range Neutron Flux
Source Range Neutron Flux

SURVEILLANCE REQUIREMENTS (continued)

Overtemperature Delta T
Overpower Delta T
Pressurizer Pressure
Pressurizer Water Level
Reactor Coolant Flow – each hot leg
RCP Bearing Water Temperature – each RCP
RCP Speed
SG Narrow Range Level – each SG
RCS Loop T-cold – each cold leg
RCS Loop T-hot – each cold leg

The Frequency is based on operating experience that demonstrates the channel failure is rare. Automated operator aids may be used to facilitate the performance of the CHANNEL CHECK.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance to the nuclear instrumentation channel output every 24 hours. If the calorimetric measurement between 70% and 100% RTP, differs from the nuclear instrument channel output by > 1% RTP, the nuclear instrument channel is not declared inoperable, but must be adjusted. If the nuclear instrument channel output cannot be properly adjusted, the channel is declared inoperable.

Three Notes modify SR 3.3.1.2. The first Note indicates that the nuclear instrument channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the nuclear instrument channel output and the calorimetric measurement between 70% and 100% RTP is > 1% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels the calorimetric data are inaccurate. The third Note is required because, at power levels between 15% and 70% calorimetric uncertainty and control rod insertion create the potential for miscalibration of the nuclear instrumentation channel in cases where the channel is adjusted downward to match the calorimetric power. Therefore, if the calorimetric heat measurement is less than 70% RTP, and if the nuclear instrumentation channel indicated power is lower than the calorimetric measurement by > 1%, then the nuclear instrumentation channel shall be adjusted upward to match the calorimetric measurement. No nuclear instrumentation channel adjustment is required if the nuclear instrumentation channel is higher than the calorimetric measurement (see Westinghouse Technical Bulletin NSD-TB-92-14, Rev. 1.)

The Frequency of every 24 hours is adequate. It is based on plant operating experience, considering instrument reliability and operating

SURVEILLANCE REQUIREMENTS (continued)

history data for instrument drift.

Together these factors demonstrate the change in the absolute difference between nuclear instrumentation and heat balance calculated powers rarely exceeds 1% RTP in any 24 hours period.

In addition, main control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the calorimetric heat balance to the calculated ΔT power ($q_{\Delta T}$) in each Division every 24 hours. If the calorimetric measurement between 70% and 100% RTP, differs from the calculated ΔT power by > 1% RTP, the Function is not declared inoperable, but the conversion factor ΔT°, must be adjusted. If ΔT° cannot be properly adjusted, the Function is declared inoperable in the affected Division(s).

Three Notes modify SR 3.3.1.3. The first Note indicates that ΔT° shall be adjusted consistent with the calorimetric results if the absolute difference between the calculated ΔT power and the calorimetric measurement between 70% and 100% RTP is > 1% RTP.

The second Note clarifies that this Surveillance is required only if reactor power is ≥ 50% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 50% RTP. At lower power levels, the calorimetric data are inaccurate. The calculated ΔT power is normally stable (less likely to need adjustment or to be grossly affected by changes in the core loading pattern than the nuclear instrumentation), and its calibration should not be unnecessarily altered by a possibly inaccurate calorimetric measurement at low power.

The third Note is required because at power levels below 70%, calorimetric uncertainty creates the potential for non-conservative adjustment of the ΔT° conversion factor, in cases where the calculated ΔT power would be reduced to match the calorimetric power. Therefore, if the calorimetric heat measurement is less than 70% RTP, and if the calculated ΔT power is lower than the calorimetric measurement by > 5%, then the ΔT° conversion factor shall be adjusted so that the calculated ΔT power matches the calorimetric measurement. No ΔT° conversion factor adjustment is required if the calculated ΔT power is higher than the calorimetric measurement.

The Frequency of every 24 hours is based on plant operating experience, considering instrument reliability and the limited effects of fuel burnup and rod position changes on the accuracy of the calculated ΔT power.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4

SR 3.3.1.4 compares the AXIAL FLUX DIFFERENCE determined using the incore system to the nuclear instrument channel AXIAL FLUX DIFFERENCE every 31 EFPD.

If the absolute difference is ≥ 3% AFD the nuclear instrument channel is still OPERABLE, but must be readjusted. If the nuclear instrument channel cannot be properly readjusted, the channel is declared inoperable. This surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT function.

Two Notes modify SR 3.3.1.4. The first Note indicates that the excore nuclear instrument channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3% AFD. Note 2 clarifies that the Surveillance is required only if reactor power is ≥ 20% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP. Below 20% RTP, the design of the incore detector system, low core power density, and detector accuracy make use of the incore detectors inadequate for use as a reference standard for comparison to the excore channels.

The Frequency of every 31 EFPD is adequate. It is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

SR 3.3.1.5

SR 3.3.1.5 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function.

SURVEILLANCE REQUIREMENTS (continued)

A Note modifies SR 3.3.1.5. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that 24 hours is allowed for performing the first surveillance after reaching 50% RTP.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.6

SR 3.3.1.6 is the performance of a TADOT every 92 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The Reactor Trip Breaker (RTB) test shall include separate verification of the undervoltage and shunt trip mechanisms. Each RTB in a division shall be tested separately in order to minimize the possibility of an inadvertent trip.

The Frequency of every 92 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data. In addition, the AP1000 design provides additional breakers to enhance reliability.

The SR is modified by a Note to clarify that both breakers in a single division are to be tested during each STAGGERED TEST.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a REACTOR TRIP CHANNEL OPERATIONAL TEST (RTCOT) every 92 days.

A RTCOT is performed on each required channel to provide reasonable assurance that the entire channel will perform the intended Function.

A test subsystem is provided with the protection and safety monitoring system to aid the plant staff in performing the RTCOT. The test subsystem is designed to allow for complete functional testing by using a combination of system self checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded.

SURVEILLANCE REQUIREMENTS (continued)

For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Generally this verification includes a comparison of the outputs from two or more redundant subsystems or channels.

Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing.

To the extent possible, protection and safety monitoring system functional testing is accomplished with continuous system self-checking features and the continuous functional testing features. The RTCOT shall include a review of the operation of the test subsystem to verify the completeness and adequacy of the results.

If the RTCOT can not be completed using the built-in test subsystem, either because of failures in the test subsystem or failures in redundant channel hardware used for functional testing, the RTCOT can be performed using portable test equipment.

This test frequency of 92 days is justified based on Reference 7 and the use of continuous diagnostic test features, such as deadman timers, cross-check of redundant channels, memory checks, numeric coprocessor checks, and tests of timers, counters and crystal time bases, which will report a failure within the protection and safety monitoring system cabinets to the operator within 10 minutes of a detectable failure.

SR 3.3.1.7 is modified by a note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.6 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for a time greater than 4 hours, this Surveillance must be performed prior to 4 hours after entry into MODE 3.

During the RTCOT, the protection and safety monitoring system cabinets in the division under test may be placed in bypass.

<u>SR 3.3.1.8</u>

SR 3.3.1.8 is the performance of a RTCOT as described in SR 3.3.1.6, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit

SURVEILLANCE REQUIREMENTS (continued)

condition. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 92 days of the Frequencies prior to reactor startup and four hours after reducing power below P-10 and P-6. The Frequency of “prior to startup” ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of “4 hours after reducing power below P-10” (applicable to intermediate and power range low channels) and “4 hours after reducing power below P-6” (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours.

<u>SR 3.3.1.9</u>

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

Transmitter calibration must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current “as found” values and the previous test “as left” values must be consistent with the transmitter drift allowance used in the setpoint methodology.

The CHANNEL CALIBRATION is assisted by the use of a tester.

SURVEILLANCE REQUIREMENTS (continued)

The setpoint methodology requires that 30 months drift be used (1.25 times the surveillance calibration interval, 24 months) based on Generic Letter 91-04, “Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle.”

SR 3.3.1.9 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

<u>SR 3.3.1.10</u>

SR 3.3.1.10 is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 20% RTP. Below 20% RTP, the design of the incore detector system, low core power density, and detector accuracy make use of the incore detectors inadequate for use as a reference standard for comparison to the excore channels. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer’s data. This Surveillance is not required for the power range detectors for entry into MODES 2 and 1, and is not required for the intermediate range detectors for entry into MODE 2, because the plant must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. 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 these components usually pass the Surveillance when performed on the 24 month Frequency.

<u>SR 3.3.1.11</u>

SR 3.3.1.11 is the performance of a TADOT of the Manual Reactor Trip, and the SI, ADS Actuation, and CMT Injection inputs from the ESF logic. This TADOT is performed every 24 months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers.

SURVEILLANCE REQUIREMENTS (continued)

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.12

This SR 3.3.1.12 verifies that the individual channel/division actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response Time testing criteria are included in Reference 2.

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping test such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements” (Ref. 10), provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

Each division response must be verified every 24 months on a STAGGERED TEST BASIS (i.e., all four Protection Channel Sets would be tested after 96 months). Response times cannot be determined during plant operation because equipment operation is required to

SURVEILLANCE REQUIREMENTS (continued)

measure response times. Experience has shown that these components usually pass this surveillance when performed on a refueling frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR 3.3.1.12 is modified by a note exempting neutron detectors from response time testing. A Note to the Surveillance indicates that neutron detectors may be excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

REFERENCES

1. Chapter 6.0, “Engineered Safety Features.”
2. Chapter 7.0, “Instrumentation and Controls.”
3. Chapter 15.0, “Accident Analysis.”
4. WCAP-16361-P, “Westinghouse Setpoint Methodology for Protection Systems – AP1000,” May 2006 (proprietary).
5. Institute of Electrical and Electronic Engineers, IEEE-603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” June 27, 1991.
6. 10 CFR 50.49, “Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants.”
7. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
8. NRC Generic Letter No. 83-27, Surveillance Intervals in Standard Technical Specifications.
9. ESBU-TB-97-01, Westinghouse Technical Bulletin, “Digital Process Rack Operability Determination Criteria,” May 1, 1997.
10. WCAP-13632-P-A (Proprietary) and WCAP-13787-A (Non-Proprietary), Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements,” January 1996.

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into distinct but interconnected modules.

Field Transmitters and Sensors

Normally, four redundant measurements using four separate sensors, are made for each variable used for actuation of ESF. The use of four channels for protection Functions is based on a minimum of two channels being required for a trip or actuation, one channel in test or bypass, and a single failure on the remaining channel. The signal selector in the Plant Control System will function correctly with only three channels. This includes two channels properly functioning and one channel having a single failure. Minimum requirements for protection and control is achieved with three channels OPERABLE. The fourth channel is provided to increase plant availability, and permits the plant to run for an indefinite time with a single channel out of service. The circuit design is able to withstand both an input failure to the control system, which may then require the protection Function actuation, and a single failure in the other channels providing the protection Function actuation. Again, a single failure will neither cause nor prevent the protection Function actuation. These requirements are described in IEEE-603 (Ref. 4). The actual number of channels provided for each plant parameter is specified in Reference 2.

Engineered Safety Features (ESF) Channel

An ESF channel extends from the sensor to the output of the associated ESF subsystem and shall include the sensor (or sensors), the signal conditioning, any associated data links, and the associated ESF subsystem. For ESF channels containing nuclear instrumentation, the ESF channel shall also include the nuclear instrument signal conditioning and the associated Nuclear Instrumentation Signal Processing and Control (NISPAC) subsystem. Any manual ESF controls that are associated with a particular ESF channel are also included in that ESF channel.

BACKGROUND (continued)

Plant Protection Subsystem

The Plant Protection contains the necessary equipment to:

- Permit acquisition and analysis of the sensor inputs, including plant process sensors and nuclear instrumentation, required for reactor trip and ESF calculations;
- Perform computation or logic operations on variables based on these inputs;
- Provide trip signals to the reactor trip switchgear and ESF actuation data to the ESF coincidence logic as required;
- Permit manual trip or bypass of each individual reactor trip Function and permit manual actuation or bypass of each individual voted ESF Function;
- Provide data to other systems in the Instrumentation and Control (I&C) architecture; and
- Provide separate input circuitry for control Functions that require input from sensors that are also required for protection Functions.

Each of the four divisions of plant protection provides signal conditioning, comparable output signals for indications in the main control room, and comparison of measured input signals with established setpoints. The basis of the setpoints are described in References 1, 2, and 3. If the measured value of a unit parameter exceeds the predetermined setpoint, an output is generated which is transmitted to the ESF coincidence logic for logic evaluation.

Within the protection and safety monitoring system, redundancy is generally provided for active equipment such as processors and communication hardware. This redundancy is provided to increase plant availability and facilitate surveillance testing. A division or channel is OPERABLE if it is capable of performing its specified safety function(s) and all the required supporting functions or systems are also capable of performing their related support functions. Thus, a division or channel is OPERABLE as long as one set of redundant components within the division or channel are capable of performing its specified safety function(s).

BACKGROUND (continued)

ESF Coincidence Logic

The ESF coincidence logic contains the necessary equipment to:

- Permit reception of the data supplied by the four divisions of plant protection and perform voting on the trip outputs;
- Perform system level logic using the input data from the plant protection subsystems and transmit the output to the ESF actuation subsystems; and
- Provide redundant hardware capable of providing system level commands to the ESF actuation subsystems.

ESF Actuation Subsystems

The ESF actuation subsystems contain the necessary equipment to:

- Receive automatic system level signals supplied by the ESF coincidence logic;
- Receive and transmit data to/from main control room multiplexers;
- Receive and transmit data to/from other PLCs on the same logic bus;
- Receive status data from component position switches (such as limit switches and torque switches); and
- Perform logic computations on received data, generate logic commands for final actuators (such as START, STOP, OPEN, and CLOSE).

ESF Coincidence Logic and ESF Actuation Subsystem OPERABILITY Background

Each ESF coincidence logic and ESF actuation subsystem has two subsystems that communicate by means of redundant halves of the logic bus. This arrangement is provided to facilitate testing. If one subsystem is removed from service, the remaining subsystem continues to function and the ESF division continues to provide full protection. At least one of these redundant halves is connected to the battery backed portion of the power system. This provides full functionality of the ESF division even when all ac power sources are lost. As long as one battery subsystem within an ESF coincidence logic or ESF actuation subsystem continues to

BACKGROUND (continued)

operate, the ESF division is unaffected. An ESF division is only affected when all battery backed subsystems within that division's ESF coincidence logic or ESF actuation subsystem are not OPERABLE.

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the trip output is set. Any trip output is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

The Trip Setpoints used in the trip output are based on the analytical limits stated in Reference 2. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.2-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the "Westinghouse Setpoint Methodology for Protection Systems" (Refs. 9 and 10). (Reference 9 is an AP600 document that describes a methodology that is applicable to AP1000. AP1000 has some slight differences in instrument spans as a result of the higher power level.) The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the trip output is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

The Trip Setpoints and Allowable Values listed in Table 3.3.2-1 are based on the methodology described in Reference 9, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

BACKGROUND (continued)

Calibration tolerances and drift allowances must be specified in plant calibration procedures, and must be consistent with the values used in the setpoint methodology.

The OPERABILITY of each transmitter or sensor can be evaluated when its “as found” calibration data are compared against the “as left” data and are shown to be within the setpoint methodology assumptions. The basis of the setpoints is described in References 1, 2, 3, and 9. Trending of transmitter calibration is required by Generic Letter 91-04, “Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle.”

The protection and safety monitoring system testing features are designed to allow for complete functional testing by using a combination of system self-checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded. For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing. To the extent possible, protection and safety monitoring system functional testing will be accomplished with continuous system self-checking features and the continuous functional testing features.

The protection and safety monitoring system incorporates continuous system self-checking features wherever practical. Self-checking features include on-line diagnostics for the computer system and the hardware and communications tests. These self-checking tests do not interfere with normal system operation.

In addition to the self-checking features, the system includes functional testing features. Functional testing features include continuous functional testing features and manually initiated functional testing features. To the extent practical functional testing features are designed not to interfere with normal system operation.

In addition to the system self-checking features and functional testing features, other test features are included for those parts of the system which are not tested with self-checking features or functional testing features. These test features allow for instruments/sensor checks,

BACKGROUND (continued)

calibration verification, response time testing, setpoint verification and component testing. The test features again include a combination of continuous testing features and manual testing features.

All of the testing features are designed so that the duration of the testing is as short as possible. Testing features are designed so that the actual logic is not modified. To prevent unwanted actuation, the testing features are designed with either the capability to bypass a Function during testing and/or limit the number of signals allowed to be placed in test at one time.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure – Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation not specifically credited in the accident safety analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These Functions may provide protection for conditions which do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3).

The LCO generally requires OPERABILITY of four channels in each instrumentation/logic Function and two devices for each manual initiation Function. The two-out-of-four configurations allow one channel to be bypassed during maintenance or testing without causing an ESFAS initiation. Two manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents. ESFAS protective functions are as follows:

1. Safeguards Actuation

   The Safeguards Actuation signal actuates the alignment of the Core Makeup Tank (CMT) valves for passive injection to the RCS. The Safeguards Actuation signal provides two primary Functions:

   - Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

heat removal and clad integrity, peak clad temperature
< 2200°F); and

- Boration to ensure recovery and maintenance of SHUTDOWN MARGIN (k_{eff} < 1.0).

These Functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The Safeguards Actuation signal is also used to initiate other Functions such as:

- Containment Isolation;
- Reactor Trip;
- Turbine Trip;
- Close Main Feedwater Control Valves;
- Trip Main Feedwater Pumps and Closure of Isolation and Crossover Valves; and
- Reactor Coolant Pump Trip.

These other Functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the turbine and reactor to limit power generation;
- Isolation of main feedwater to limit secondary side mass losses;
- Trip of the reactor coolant pumps to ensure proper CMT actuation;
- Enabling automatic depressurization of the RCS on CMT Level – Low 1 to ensure continued safeguards actuated injection.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Manual and automatic initiation of Safeguards Actuation must be OPERABLE in MODES 1, 2, 3, and 4. In these MODES there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Automatic actuation in MODE 4 is provided by the high containment pressure signal.

Manual initiation is required in MODE 5 to support system level initiation. Automatic initiation is not required to be OPERABLE in MODE 5 because parameters are not available to provide automatic actuation, and manual initiation is sufficient to mitigate the consequences of an accident.

These Safeguards Actuation Functions are not required to be OPERABLE in MODE 6 because there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems.

1.a. Manual Initiation

The LCO requires that two manual initiation devices are OPERABLE. The operator can initiate the Safeguards Actuation signal at any time by using either of two switches in the main control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO on Manual Initiation ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each device consists of one switch and the interconnecting wiring to all four divisions. Each manual initiation device actuates all four divisions. This configuration does not allow testing at power.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

1.b. Containment Pressure – High 2

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

The transmitters (d/p cells) and electronics are located inside of containment. Since the transmitters and electronics are located inside of containment, they will experience adverse environmental conditions and the trip setpoint reflects environmental instrument uncertainties. The Containment Pressure – High 2 setpoint has been specified as low as reasonable, without creating potential for spurious trips during normal operations, consistent with the TMI action item (NUREG-0933, Item II.E.4.2) guidance.

The LCO requires four channels of Containment Pressure – High 2 to be OPERABLE in MODES 1, 2, 3, and 4. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

1.c. Pressurizer Pressure – Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) safety valve;
- SLB;
- A spectrum of rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer safety valve;
- LOCAs; and
- Steam Generator Tube Rupture (SGTR).

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

The LCO requires four channels of Pressurizer Pressure – Low to be OPERABLE in MODES 1, 2, and 3 (above P-11, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F), to mitigate the consequences of a high energy line rupture inside containment. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic actuation below this pressure is then performed by the Containment Pressure – High 2 signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF Functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

1.d. <u>Steam Line Pressure – Low</u>

Steam Line Pressure – Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

It is possible for the transmitters to experience adverse environmental conditions during a secondary side break. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

This Function is anticipatory in nature and has a typical lead/lag ratio of 50/5.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The LCO requires four channels of Steam Line Pressure – Low to be OPERABLE in MODES 1, 2, and 3 (above P-11, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F). At these conditions, a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. Four channels are provided in each steam line to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern, inside containment SLB will be terminated by automatic actuation via Containment Pressure – High 2, and outside containment SLB will be terminated by the Steam Line Pressure-Negative Rate – High signal for steam line isolation. In MODE 4, 5, or 6, this Function is not needed for accident detection and mitigation because the steam line pressure is below the actuation setpoint. Low steam line pressure in these MODES is not an adequate indication of a feed line or steam line break.

1.e. RCS Cold Leg Temperature (T_{cold}) – Low

This signal provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

The LCO requires four channels of T_{cold} – Low to be OPERABLE in MODES 1 and 2, and in MODE 3 with any main steam isolation valve open and above P-11 when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F. At these conditions, a secondary side break or stuck open valve could result in the rapid cooldown of the primary side. Four channels are provided in each loop to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation because the cold leg temperature is reduced below the actuation setpoint.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

2. Core Makeup Tank (CMT) Actuation

CMT Actuation provides the passive injection of borated water into the RCS. Injection provides RCS makeup water and boration during transients or accidents when the normal makeup supply from the Chemical and Volume Control System (CVS) is lost or insufficient. Two tanks are available to provide passive injection of borated water. CMT injection mitigates the effects of high energy line breaks by adding primary side water to ensure maintenance or recovery of reactor vessel water level following a LOCA, and by borating to ensure recovery or maintenance of SHUTDOWN MARGIN following a steam line break. CMT Valve Actuation is initiated by the Safeguards Actuation signal, Pressurizer Level – Low 2, ADS Stages 1, 2 and 3 Actuation, or manually.

The LCO requires that manual and automatic CMT Valve Actuation be OPERABLE in MODES 1 through 4. Manual and Automatic actuation of the CMT valves is additionally required in MODE 5 with the RCS pressure boundary intact. Actuation of this Function is not required in MODE 5 with the RCS pressure boundary open, or MODE 6 because the CMTs are not required to be OPERABLE in these MODES.

2.a. Manual Initiation

Manual CMT Valve Actuation is accomplished by either of two switches in the main control room. Either switch activates all four divisions.

2.b. Pressurizer Water Level – Low 2

This Function also initiates CMT Valve Actuation from the coincidence of pressurizer level below the Low 2 Setpoint in any two of the four divisions. This function can be manually blocked when the pressurizer water level is below the P-12 Setpoint. This Function is automatically unblocked when the pressurizer water level is above the P-12 Setpoint. The Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide protection for an event that results in a harsh environment.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

2.c. Safeguards Actuation (Function 1)

CMT Valve Actuation is also initiated by all Functions that initiate the Safeguards Actuation signal. The CMT Valve Actuation Function requirements are the same as the requirements for the Safeguards Actuation Functions, but only apply in MODES 1 through 4, and in MODE 5 with the RCS pressure boundary intact. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1 is referenced for all initiating Functions and requirements.

2.d. ADS Stages 1, 2, and 3 Actuation (Function 9)

The CMTs are actuated on an ADS Stages 1, 2, and 3 actuation. The CMT Actuation Function requirements are the same as the requirements for the ADS Stages 1, 2, and 3 Actuation Function, but only apply in MODES 1 through 4, and in MODE 5 with the RCS pressure boundary intact. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 9 is referenced for all initiating functions and requirements.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere and selected process systems which penetrate containment from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

Containment Isolation is actuated by the Safeguards Actuation signal, manual actuation of containment cooling, or manually.

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Manual and automatic initiation of Containment Isolation must be OPERABLE in MODES 1, 2, 3, and 4, when containment integrity is required. Manual initiation is required in MODE 5 and MODE 6 for closure of open penetrations providing direct access from the containment atmosphere to the outside atmosphere. Manual initiation of this Function in MODES 5 and 6 is not applicable if the direct access lines penetrating containment are isolated. Initiation of containment isolation by manual initiation of passive containment cooling in MODE 5 or 6 with decay heat ≤ 9.0 MWt is not required because OPERABILITY of the passive containment cooling system is not required when air cooling is sufficient. This provides the capability to manually initiate containment isolation during all MODES. Automatic Safeguards Actuation is required in MODE 5 for closure of open penetrations providing direct access from the containment atmosphere to the outside atmosphere. Automatic Safeguards Actuation is not required in MODE 6 because manual initiation is sufficient to mitigate the consequences of an accident in this MODE.

3.a. Manual Initiation

Manual Containment Isolation is accomplished by either of two switches in the main control room. Either switch actuates all four ESFAC divisions.

3.b. Manual Initiation of Passive Containment Cooling (Function 12.a)

Containment Isolation is also initiated by Manual Initiation of Passive Containment Cooling. This is accomplished as described for ESFAS Function 12.a, but are not applicable if the direct access flow paths are isolated.

3.c. Safeguards Actuation (Function 1)

Containment Isolation is also initiated by all Functions that initiate the Safeguards Actuation signal. The Containment Isolation Function requirements are the same as the requirements for the Safeguards Actuation Function, but are not applicable if the direct access flow paths are isolated. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1 is referenced for all initiating functions and requirements.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam

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lines will limit the steam break accident to the blowdown from one SG at most. For an SLB upstream of the isolation valves, inside or outside of containment, closure of the isolation valves limits the accident to the blowdown from only the affected SG. For a SLB downstream of the isolation valves, closure of the isolation valves terminates the accident as soon as the steam lines depressurize.

Closure of the turbine stop and control valves and the main steam branch isolation valves is initiated by this Function. Closure of these valves limits the accidental depressurization of the main steam system associated with an inadvertent opening of a single steam dump, relief, safety valve, or a rupture of a main steam line. Closure of these valves also supports a steam generator tube rupture event by isolating the faulted steam generator.

4.a. Manual Initiation

Manual initiation of Steam Line Isolation can be accomplished from the main control room. There are two switches in the main control room and either switch can initiate action to immediately close all main steam isolation valves (MSIVs). The LCO requires two OPERABLE channels in MODES 1, 2, 3, and 4 with any main steam valve open, when there is sufficient energy in the RCS and SGs to have an SLB or other accident resulting in the release of significant quantities of energy to cause a cooldown of the primary system. In MODES 5 and 6, this Function is not required to be OPERABLE because there is insufficient energy in the secondary side of the unit to cause an accident.

4.b. Containment Pressure – High 2

This Function actuates closure of the MSIVs in the event a SLB inside containment to limit the mass and energy release to containment and limit blowdown to a single SG.

The transmitters and electronics are located inside containment, thus, they will experience harsh environmental conditions and the Trip Setpoint reflects environmental instrument uncertainties.

The Containment Pressure – High 2 setpoint has been specified as low as reasonable, without creating potential for spurious trips during normal operations, consistent with the TMI action item (NUREG-0933, Item II.E.4.2) guidance. The LCO requires four channels of Containment Pressure – High 2 to be OPERABLE in MODES 1, 2, 3, and 4, with any main

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

steam valve open, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure

no single random failure will disable this trip Function. There would be a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. In MODES 5 and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure – High 2 setpoint.

4.c. Steam Line Pressure

(1) Steam Line Pressure – Low

Steam Line Pressure – Low provides closure of the MSIVs in the event of an SLB to limit the mass and energy release to containment and limit blowdown to a single SG.

The LCO requires four channels of Steam Line Pressure – Low Function to be OPERABLE in MODES 1, 2, and 3 (above P-11, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F), with any main steam isolation valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. Four channels are provided in each steam line to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure – High 2, and stuck open valve transients and outside containment steam line breaks will be terminated by the Steam Line Pressure-Negative Rate – High signal for Steam Line Isolation. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

(2) Steam Line Pressure-Negative Rate – High

Steam Line Pressure-Negative Rate – High provides closure of the MSIVs for an SLB, when less than the P-11 setpoint, to maintain at least one unfaulted SG as a

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

heat sink for the reactor and to limit the mass and energy release to containment. When the operator manually

blocks the Steam Line Pressure – Low when less than the P-11 setpoint, the Steam Line Pressure-Negative Rate – High signal is automatically enabled.

The LCO requires four channels of Steam Line Pressure-Negative Rate – High to be OPERABLE in MODE 3, with any main steam valve open, when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). Four channels are provided in each steam line to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODES 1 and 2, and in MODE 3 when above the P-11 setpoint with the RCS boron concentration below that necessary to meet the SDM requirements at an RCS temperature of 200°F, this signal is automatically disabled and the Steam Line Pressure – Low signal is automatically enabled.

In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

While the transmitters may experience elevated ambient temperatures due to a steam line break, the Trip Function is on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

4.d. T_{cold} – Low

This Function provides closure of the MSIVs during a SLB or inadvertent opening of a SG relief or a safety valve to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment.

This Function was discussed as Safeguards Actuation Function 1.e.

The LCO requires four channels of T_{cold} – Low to be OPERABLE in MODES 1 and 2, and in MODE 3 above P-11 when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F, with any main steam isolation valve open, when a secondary

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side break or stuck open valve could result in the rapid

cooldown of the primary side. Four channels are provided in each loop to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODE 3 below P-11 and in MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation because the cold leg temperature is reduced below the actuation setpoint.

5. Turbine Trip

The primary Function of the Turbine Trip is to prevent damage to the turbine due to water in the steam lines. This Function is necessary in MODES 1 and 2, and 3 above P-11 to mitigate the effects of a large SLB or a large Feedline Break (FLB). Failure to trip the turbine following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment. In MODES 3, 4, 5, and 6, the turbine is not in operation and this function is not required to be OPERABLE.

This Function is actuated by Steam Generator Water Level – High 2, by a Safeguards Actuation signal, or manually. The Reactor Trip Signal also initiates a turbine trip signal whenever a reactor trip (P-4) is generated.

5.a. Manual Main Feedwater Isolation

The Turbine Trip is also initiated by the Manual Main Feedwater Control Valve Isolation Function. The requirements for this Function are the same as the requirements for Manual Main Feedwater Control Valve Isolation (Function 6.a), but only apply in MODES 1 and 2. Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 6.a is referenced for all requirements.

5.b. Steam Generator Narrow Range Water Level – High 2

This signal provides protection against excessive feedwater flow by closing the main feedwater control, isolation and crossover valves, tripping of the main feedwater pumps, and tripping the turbine. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The transmitters (d/p cells) are located inside containment.

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However, the events which this Function protect against cannot cause severe environment in containment. Therefore, the Setpoint reflects only steady state instrument uncertainties.

5.c. Safeguards Actuation (Function 1)

Turbine Trip is also initiated by all Functions that initiate the Safeguards Actuation signal. The Turbine Trip Function requirements are the same as the requirements for the Safeguards Actuation Function, but only apply in MODES 1 and 2. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1 is referenced for all initiating Functions and requirements. The Safeguards Actuation signal closes all main feedwater control, isolation and crossover valves, trips all main feedwater pumps, and trips the turbine.

5.d. Reactor Trip (Function 18.a)

Turbine Trip is also initiated by all functions that initiate Reactor Trip. The turbine trip function requirements are the same as the requirements for the Reactor Trip Function, but only apply in MODES 1 and 2. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 18.a, P-4 (Reactor Trip), is referenced for all initiating Functions and requirements.

6. Main Feedwater Control Valve Isolation

The primary Function of Main Feedwater Control Valve Isolation is to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the SGs. This Function is actuated by Steam Generator Narrow Range Water Level – High 2, by a Safeguards Actuation signal, or manually. The Reactor Trip Signal also initiates closure of the main feedwater control valves coincident with a low RCS average temperature (T_{avg}) signal whenever a reactor trip (P-4) is generated.

Closing the Main Feedwater Control Valves on Manual Main Feedwater Isolation, SG Narrow Range Water Level-High 2, or Safeguards Actuation is necessary in MODES 1, 2, and 3 to mitigate the effects of a large SLB or a large FLB. This Function is also required to be OPERABLE in MODES 1 and 2 on T_{avg} Low-1 coincident with Reactor Trip (P-4). Failure to close the main feedwater control valves following a SLB or FLB can lead to

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment. Manual main feedwater isolation is required to be OPERABLE in MODE 4 when the main feedwater control valves are open. This Function is not applicable in MODE 4 for valve isolation if the main feedwater line is isolated. Automatic actuation on a Steam Generator Narrow Range Water Level – High 2 is required to be OPERABLE in MODE 4 when the RCS is not being cooled by the RNS. In MODES 5 and 6, the energy in the RCS and the steam generators is low and this function is not required to be OPERABLE.

6.a. Manual Main Feedwater Isolation

Manual Main Feedwater Isolation can be accomplished from the main control room. There are two switches in the main control room and either switch can initiate action in both divisions to close all main and startup feedwater control, isolation and crossover valves, trip all main and startup feedwater pumps, and trip the turbine.

6.b. Steam Generator Narrow Range Water Level – High 2

This signal provides protection against excessive feedwater flow by closing the main feedwater control, isolation and crossover valves, tripping of the Main Feedwater Pumps, and tripping the turbine.

Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The transmitters (d/p cells) are located inside containment. However, the events which this Function protect against cannot cause severe environment in containment. Therefore, the Setpoint reflects only steady state instrument uncertainties.

6.c. Safeguards Actuation (Function 1)

This Function is also initiated by all Functions that initiate the Safeguards Actuation signal. The Main Feedwater Control Valve Isolation Function requirements are the same as the requirements for the Safeguards Actuation Function, but do not apply in MODE 4 with the flow paths isolated. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead

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Function 1 is referenced for all initiating Functions and requirements. The Safeguards Actuation signal closes all main feedwater control, isolation and crossover valves, trips all main feedwater pumps, and trips the turbine.

6.d. T_{avg} Low-1 Coincident with Reactor Trip (P-4)

This signal provides protection against excessive feedwater flow by closing the main feedwater control valves. This signal results from a coincidence of two of the four divisions of reactor loop average temperature below the Low 1 setpoint coincident with the P-4 permissive. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this trip Function.

7. Main Feedwater Pump Trip and Valve Isolation

The primary function of the Main Feedwater Pump Trip and Isolation is to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the SGs. Valve isolation includes closing the main feedwater isolation and crossover valves. Isolation of main feedwater is necessary to prevent an increase in heat removal from the reactor coolant system in the event of a feedwater system malfunction. Addition of excessive feedwater causes an increase in core power by decreasing reactor coolant temperature. This Function is actuated by Steam Generator Water Level – High 2, by a Safeguards Actuation signal, or manually. The Reactor Trip Signal also initiates a turbine trip signal whenever a reactor trip (P-4) is generated.

This Function is necessary in MODES 1, 2, 3, and 4 to mitigate the effects of a large SLB or a large FLB except T_{avg} Low 2 coincident with Reactor Trip (P-4) which is required to be OPERABLE in MODES 1 and 2. Failure to trip the turbine or isolate the main feedwater system following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment. Manual main feedwater isolation is required to be OPERABLE in MODE 4 when the main feedwater isolation valves are open. This Function is not applicable in MODE 4 for valve isolation if the main feedwater line is isolated. Automatic actuation on a Steam Generator Narrow Range Water Level – High 2 is required to be OPERABLE in MODE 4 when the RCS is not being

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cooled by the RNS. In MODES 5 and 6, the energy in the RCS and the steam generators is low and this Function is not required to be OPERABLE.

7.a. Manual Main Feedwater Isolation

The Main Feedwater Pump Trip and Valve Isolation is also initiated by the Manual Main Feedwater Control Valve Isolation Function. The requirements for this Function are the same as the requirements for Manual Main Feedwater Control Valve Isolation (Function 6.a). Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 6.a is referenced for all requirements.

7.b. Steam Generator Narrow Range Water Level – High 2

This signal provides protection against excessive feedwater flow by closing the main feedwater control, isolation and crossover valves, tripping of the main feedwater pumps, and tripping the turbine. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The transmitters (d/p cells) are located inside containment. However, the events which this Function protect against cannot cause severe environment in containment. Therefore, the Setpoint reflects only steady state instrument uncertainties.

7.c. Safeguards Actuation (Function 1)

This Function is also initiated by all Functions that initiate the Safeguards Actuation signal. The Main Feedwater Pump Trip and Valve Isolation Function requirements are the same as the requirements for their Safeguards Actuation Function, but do not apply in MODE 4 with the flow paths isolated. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1 is referenced for all initiating Functions and requirements. The Safeguards Actuation signal closes all main feedwater control, isolation and crossover valves, trips all main feedwater pumps, and trips the turbine.

7.d. T_{avg} Low-2 Coincident with Reactor Trip (P-4)

This signal provides protection against excessive feedwater flow by closing the main feedwater isolation and crossover leg valves, and tripping of the main feedwater pumps. This signal

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results from a coincidence of two out of four divisions of reactor loop average temperature below the Low 2 setpoint coincident with the P-4 permissive. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this trip Function. This Function may be manually blocked when the pressurizer pressure is below the P-11 setpoint. The block is automatically removed when the pressurizer pressure is above the P-11 setpoint.

8. Startup Feedwater Isolation

The primary Function of the Startup Feedwater Isolation is to stop the excessive flow of feedwater into the SGs. This Function is necessary in MODES 1, 2, 3, and 4 to mitigate the effects of a large SLB or a large FLB. Failure to isolate the startup feedwater system following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment.

Startup feedwater isolation must be OPERABLE in MODES 1, 2, 3, and 4 when there is significant mass and energy in the RCS and the steam generators. This Function is not applicable in MODE 4 when the startup feedwater flow paths are isolated. In MODES 5 and 6, the energy in the RCS and the steam generators is low and this Function is not required to be OPERABLE.

8.a. Steam Generator (SG) Narrow Range Water Level – High 2

If steam generator narrow range level reaches the High 2 setpoint in either steam generator, then all startup feedwater control and isolation valves are closed and the startup feedwater pumps are tripped. Four channels are provided in each steam generator to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

8.b. T_{cold} – Low

This Function closes the startup feedwater control and isolation valves and trips the startup feedwater pumps if reactor coolant system cold leg temperature is below the T_{cold} setpoint in any loop. Startup feedwater isolation on this condition may be manually blocked when the pressurizer

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pressure is below the P-11 setpoint. This function is automatically unblocked when the pressurizer pressure is above the P-11 setpoint with the RCS boron concentration below that necessary to meet the SDM requirements at an RCS temperature of 200°F. Four channels are provided in each loop to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

8.c. Manual Main Feedwater Control Valve Isolation (Function 6.a)

The Startup Feedwater Isolation is also initiated by the Manual Main Feedwater Control Valve Isolation Function. The requirements for this Function are the same as the requirements for the Manual Main Feedwater Control Valve Isolation (Function 6.a). Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 6.a is referenced for all requirements.

9. ADS Stages 1, 2, & 3 Actuation

The Automatic Depressurization System (ADS) provides a sequenced depressurization of the reactor coolant system to allow passive injection from the CMTs, accumulators, and the in-containment refueling water storage tank (IRWST) to mitigate the effects of a LOCA. The depressurization is accomplished in four stages, with the first three stages discharging into the IRWST and the last stage discharging into containment. Each of the first three stages consists of two parallel paths with each path containing an isolation valve and a depressurization valve.

The first stage isolation valves open on any ADS Stages 1, 2, and 3 actuation. The first stage depressurization valves are opened following a preset time delay after the actuation of the isolation valves. The second stage isolation valves are opened following a preset time delay after actuation of the first stage depressurization valves open. The second stage depressurization valves are opened following a preset time delay after the second stage isolation valves are actuated, similar to stage one. Similar to the second stage, the third stage isolation valves are opened following a preset time delay after the actuation of the second stage depressurization valves. The third stage depressurization valves are opened following a preset time delay after the third stage isolation valves are actuated.

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9.a. Manual Initiation

The first stage depressurization valves open on manual actuation. Any ADS Stages 1, 2, and 3 actuation also actuates PRHR and trips all reactor coolant pumps. The operator can initiate an ADS Stages 1, 2, and 3 actuation from the main control room by simultaneously actuating two ADS actuation devices in the same set. There are two sets of two switches each in the main control room. Simultaneously actuating the two devices in either set will actuate ADS Stages 1, 2, and 3. This Function must be OPERABLE in MODES 1, 2, 3, and 4. This Function must also be OPERABLE in MODES 5 and 6 when the required ADS valves are not open, and in MODE 6 with the upper internals in place. The required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS - Shutdown, RCS Intact and LCO 3.4.13, ADS - Shutdown, RCS Open.

9.b. CMT Level – Low 1 Coincident with CMT Actuation

This Function ensures continued passive injection or borated water to the RCS following a small break LOCA. ADS Stages 1, 2 and 3 actuation is initiated when the CMT Level reaches its Low 1 Setpoint coincident with any CMT Actuation signal (Function 2). Four channels are provided in each CMT to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

The ADS Stages 1, 2, and 3 Actuation Function requirements are the same as the requirements discussed in Function 2 (CMT Actuation). Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 2 is referenced for all initiating functions and requirements. This Function must be OPERABLE in MODES 1, 2, 3, and 4.

This Function must also be OPERABLE in MODE 5 with pressurizer level ≥ 20% and the required ADS valves not open. The required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS - Shutdown, RCS Intact and LCO 3.4.13, ADS - Shutdown, RCS Open. In MODE 5, only one CMT is required to be OPERABLE in accordance with LCO 3.5.3, CMTs - Shutdown, RCS Intact; therefore, CMT level channels are only required on an OPERABLE CMT.

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10. ADS Stage 4 Actuation

The ADS provides a sequenced depressurization of the reactor coolant system to allow passive injection from the CMTs, accumulators, and the IRWST to mitigate the effects of a LOCA. The depressurization is accomplished in four stages, with the first three stages discharging into the IRWST and the fourth stage discharging into containment.

The fourth stage of the ADS consists of four parallel paths. Each of these paths consists of a normally open isolation valve and a depressurization valve. The four paths are divided into two groups with two paths in each group. Within each group, one path is designated to be substage A and the second path is designated to be substage B.

The substage A depressurization valves are opened following a preset time delay after the substage A isolation valve confirmatory open signal. The sequence is continued with substage B. A confirmatory open signal is provided to the substage B isolation valves following a preset time delay after the substage A depressurization valve has been opened. The signal to open the substage B depressurization valve is provided following a preset time delay after the substage B isolation valves confirmatory open signal.

10.a. Manual Initiation Coincident with RCS Wide Range Pressure – Low or ADS Stages 1, 2, and 3 Actuation (Function 9)

The fourth stage depressurization valves open on manual actuation. The operator can initiate Stage 4 of ADS from the main control room. There are two sets of two switches each in the main control room. Actuating the two switches in either set will actuate all 4th stage ADS valves. This manual actuation is interlocked to actuate with either the low RCS pressure signal or with the ADS Stages 1, 2, & 3 actuation (Function 9). These interlocks minimize the potential for inadvertent actuation of this Function. This interlock with Function 9 allows manual actuation of this Function if automatic or manual actuation of the ADS Stages 1, 2, & 3 valves fails to depressurize the RCS due to common-mode failure. This consideration is important in PRA modeling to improve the reliability of reducing the RCS pressure following a small LOCA or transient event. This Function must be OPERABLE

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in MODES 1, 2, 3, and 4. This Function must also be OPERABLE in MODES 5 and 6 when the required ADS valves are not open, and in MODE 6 with the upper internals in place. The required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS - Shutdown, RCS Intact and LCO 3.4.13, ADS - Shutdown, RCS Open.

10.b. CMT Level – Low 2 Coincident with RCS Wide Range Pressure – Low

The fourth stage depressurization valves open on CMT Level – Low 2 in two-out-of-four channels in either CMT. Actuation of the fourth stage depressurization valves is interlocked with the third stage depressurization signal such that the fourth stage is not actuated unless the third stage has been previously actuated following a preset time delay. Actuation of the fourth stage ADS valves are further interlocked with a low RCS pressure signal such that the ADS Stage 4 actuation is not actuated unless the RCS pressure is below a predetermined setpoint. Four channels of CMT level are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This Function must be OPERABLE in MODES 1, 2, 3, and 4. This Function must also be OPERABLE in MODE 5 when the required ADS valves are not open and with the pressurizer level ≥ 20%. The required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS - Shutdown, RCS Intact and LCO 3.4.13, ADS - Shutdown, RCS Open. In MODE 5, only one CMT is required to be OPERABLE in accordance with LCO 3.5.3, CMTs - Shutdown, RCS Intact; therefore, CMT level channels are only required on an OPERABLE CMT.

10.c. Coincident RCS Loop 1 and 2 Hot Leg Level – Low

A signal to automatically open the ADS Stage 4 is also generated when coincident loop 1 and 2 reactor coolant system hot leg level indication decreases below an established setpoint for a duration exceeding an adjustable time delay. This Function is required to be OPERABLE in MODE 4 with the RCS being cooled by the RNS. This Function is also required to be OPERABLE in MODE 5 and in MODE 6 when the required ADS valves are not open. The

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required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS - Shutdown, RCS Intact and LCO 3.4.13, ADS - Shutdown, RCS Open.

11. Reactor Coolant Pump Trip

Reactor Coolant Pump (RCP) Trip allows the passive injection of borated water into the RCS. Injection provides RCS makeup water and boration during transients or accidents when the normal makeup supply from the CVS is lost or insufficient. Two tanks provide passive injection of borated water by gravity when the reactor coolant pumps are tripped. CMT injection mitigates the effects of high energy line breaks by adding primary side water to ensure maintenance or recovery of reactor vessel water level following a LOCA, and by borating to ensure recovery or maintenance of SHUTDOWN MARGIN following a steam line break. RCP trip on high bearing water temperature protects the RCP coast down. A high bearing water temperature trip signal will result in the tripping of all the RCPs. RCP trip is actuated by High RCP bearing water temperature, ADS Stages 1, 2, and 3 Actuation (Function 9), Manual CMT Actuation (Function 2.a), Pressurizer Water Level – Low 2, and Safeguards Actuation (Function 1).

11.a. ADS Stage 1, 2, and 3 Actuation (Function 9)

The RCPs are tripped any time ADS Stage 1, 2, and 3 actuation is initiated. The RCP trip Function requirements for the ADS Stage 1, 2, and 3 actuation are the same as the requirements for the ADS Function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 9 is referenced for all initiating functions and requirements.

11.b. Reactor Coolant Pump Bearing Water Temperature – High

The RCPs are tripped if two-out-of-four sensors on any RCP indicate high bearing water temperature. This Function is required to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

11.c. Manual CMT Actuation (Function 2.a)

RCP trip is also initiated by the manual CMT actuation Function. The RCP trip Function requirements are the same

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

as the requirements for the manual CMT actuation Function. Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 2.a is referenced for all requirements.

11.d. Pressurizer Water Level – Low 2

The RCPs are tripped when the pressurizer water level reaches its Low 2 setpoint. This signal results from the coincidence of pressurizer water level below the Low 2 setpoint in any two-of-four divisions. This Function is required to be OPERABLE in MODES 1, 2, 3, and 4. This Function is also required to be OPERABLE in MODE 5 with pressurizer level ≥ 20%, when the RCS is not being cooled by the RNS. This Function can be manually blocked when the pressurizer water level is below the P-12 setpoint. This Function is automatically unblocked when the pressurizer water level is above the P-12 setpoint.

11.e. Safeguards Actuation (Function 1)

This Function is also initiated by all Functions that initiated the Safeguards Actuation signal. The requirements for the reactor trip Functions are the same as the requirements for the Safeguards Actuation Function. Therefore, the requirements are not repeated in Table 3.3.2.1. Instead, Function 1 is referenced for all initiating Functions and requirements.

12. Passive Containment Cooling Actuation

The Passive Containment Cooling System (PCS) transfers heat from the reactor containment to the environment. This Function is necessary to prevent the containment design pressure and temperature from being exceeded following any postulated DBA (such as LOCA or SLB). Heat removal is initiated automatically in response to a Containment Pressure – High 2 signal or manually.

A Passive Containment Cooling Actuation signal initiates water flow by gravity by opening the isolation valves. The water flows onto the containment dome, wetting the outer surface. The path for natural circulation of air along the outside walls of the containment structure is always open.

The LCO requires this Function to be OPERABLE in MODES 1, 2, 3, and 4 when the potential exists for a DBA that could require the

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

operation of the Passive Containment Cooling System. In MODES 5 and 6, with decay heat more than 9.0 MWt, manual initiation of the PCS provides containment heat removal. Section B 3.6.7, Applicability, provides the basis for the decay heat limit.

12.a. Manual Initiation

The operator can initiate Containment Cooling at any time from the main control room by actuating either of the two containment cooling actuation switches. There are two switches in the main control room, either of which will actuate containment cooling in all divisions. Manual Initiation of containment cooling also actuates containment isolation.

12.b. Containment Pressure – High 2

This signal provides protection against a LOCA or SLB inside containment. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

The transmitters and electronics are located inside containment, thus, they will experience harsh environmental conditions and the trip setpoint reflects only steady state instrument uncertainties associated with the containment environment. The Containment Pressure – High 2 setpoint has been specified as low as reasonable, without creating potential for spurious trips during normal operations, consistent with the TMI action item (NUREG-0933, Item II.E.4.2) guidance.

13. PRHR Heat Exchanger Actuation

The PRHR Heat Exchanger (HX) provides emergency core decay heat removal when the Startup Feedwater System is not available to provide a heat sink. PRHR is actuated when the discharge valves are opened in response to Steam Generator Narrow Range (NR) Level – Low coincident with Startup Feedwater Flow – Low, Steam Generator Wide Range (WR) Level – Low, ADS Stages 1, 2, and 3 Actuation, CMT Actuation, Pressurizer Water Level – High 3, or Manual Initiation.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

13.a. Manual Initiation

Manual PRHR actuation is accomplished by either of two switches in the main control room. Either switch actuates all four ESFAC Divisions.

This Function is required to be OPERABLE in MODES 1, 2, 3, and 4, and MODE 5 with the RCS pressure boundary intact. This ensures that PRHR can be actuated in the event of a loss of the normal heat removal systems.

13.b. Steam Generator Narrow Range Level – Low Coincident with Startup Feedwater Flow – Low

PRHR is actuated when the Steam Generator Narrow Range Level reaches its low setpoint coincident with an indication of low Startup Feedwater Flow.

The LCO requires four channels per steam generator to be OPERABLE to satisfy the requirements with a two-out-of-four logic. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide protection for an event that results in a harsh environment.

Startup Feedwater Flow – Low uses a one-out-of-two logic on each of the two startup feedwater lines. This Function is required to be OPERABLE in MODES 1, 2, and 3 and in MODE 4 when the RCS is not being cooled by the Normal Residual Heat Removal System (RNS). This ensures that PRHR can be actuated in the event of a loss of the normal heat removal systems. In MODE 4 when the RCS is being cooled by the RNS, and in MODES 5 and 6, the SGs are not required to provide the normal RCS heat sink. Therefore, startup feedwater flow is not required, and PRHR actuation on low startup feedwater flow is not required.

13.c. Steam Generator Wide Range Level – Low

PRHR is also actuated when the SG Wide Range Level reaches its Low Setpoint. There are four wide range level channels for each steam generator and a two-out-of-four logic

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

is used. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This Function is required to be OPERABLE in MODES 1, 2, and 3 and in MODE 4 when the RCS is not being cooled by the RNS. This ensures that PRHR can be actuated in the event of a loss of the normal heat removal systems. In MODE 4 when the RCS is being cooled by the RNS, and in MODES 5 and 6, the SGs are not required to provide the normal RCS heat sink. Therefore, SG Wide Range Level is not required, and PRHR actuation on low wide range SG level is not required.

13.d. ADS Stages 1, 2, and 3 Actuation

PRHR is also actuated any time ADS Stages 1, 2, and 3 Actuation is initiated. The PRHR actuation Function requirements for the ADS Stages 1, 2, and 3 actuation are the same as the requirements for the ADS Stages 1, 2, and 3 Actuation Function, but only in MODES 2, 3, and 4, and in MODE 5 with the RCS pressure boundary intact.

13.e. CMT Actuation (Function 2)

PRHR is also actuated by all the Functions that actuate CMT injection. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 2 (CMT Actuation) is referenced for all initiating functions and requirements.

13.f. Pressurizer Water Level – High 3

PRHR is actuated when the pressurizer water level reaches its High 3 setpoint. This signal provides protection against a pressurizer overfill following an inadvertent core makeup tank actuation with consequential loss of offsite power. This Function is automatically unblocked when RCS pressure is above the P-19 setpoint. This Function is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the RCS is not being cooled by the RNS and above the P-19 (RCS pressure) interlock. This Function is not required to be OPERABLE in MODES 5 and 6 because it is not required to mitigate DBA in these MODES.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

14. Steam Generator Blowdown Isolation

The primary Function of the steam generator blowdown isolation is to ensure that sufficient water inventory is present in the steam generators to remove the excess heat being generated until the decay heat has decreased to within the PRHR HX capability.

This Function closes the isolation valves of the Steam Generator Blowdown System in both steam generators when a signal is generated from the PRHR HX Actuation or Steam Generator Narrow Range Water Level – Low. This Function is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the RCS is not being cooled by the RNS. This Function is not required to be OPERABLE in MODE 4 if the steam generator blowdown line is isolated.

14.a. PRHR Heat Exchanger Actuation (Function 13)

Steam Generator Blowdown Isolation is also initiated by all Functions that initiate PRHR actuation. The Steam Generator Blowdown Isolation requirements for these Functions are the same as the requirements for the PRHR Actuation. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 13, PRHR HX Actuation, is referenced for all initiating Functions and requirements.

14.b. Steam Generator Narrow Range Level – Low

The Steam Generator Blowdown isolation is actuated when the Steam Generator Narrow Range Level reaches its Low Setpoint.

The LCO requires four channels per steam generator to be OPERABLE to satisfy the requirements with a two-out-of-four logic. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide protection for an event that results in a harsh environment.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

15. Boron Dilution Block

The block of boron dilution is accomplished by closing the CVS suction valves to demineralized water storage tanks, and aligning the boric acid tank to the CVS makeup pumps. This Function is actuated by Source Range Neutron Flux Doubling and Reactor Trip.

15.a. Source Range Neutron Flux Doubling

A signal to block boron dilution in MODES 2 or 3, when not critical or during an intentional approach to criticality, and MODES 4 or 5 is derived from source range neutron flow increasing at an excessive rate (source range flux doubling). This Function is not applicable in MODES 4 and 5 if the demineralized water makeup flowpath is isolated. The source range neutron detectors are used for this Function. The LCO requires four divisions to be OPERABLE. There are four divisions and two-out-of-four logic is used. On a coincidence of excessively increasing source range neutron flux in two of the four divisions, demineralized water is isolated from the makeup pumps and reactor coolant makeup is isolated from the reactor coolant system to preclude a boron dilution event. In MODE 6, a dilution event is precluded by the requirement in LCO 3.9.2 to close, lock and secure at least one valve in each unborated water source flow path.

15.b. Reactor Trip (Function 18.a)

Demineralized Water Makeup is also isolated by all the Functions that initiate a Reactor Trip. The isolation requirements for these Functions are the same as the requirements for the Reactor Trip Function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 18.a, (P-4 Reactor Trip Breakers), is referenced for all initiating Functions and requirements.

16. Chemical Volume and Control System Makeup Line Isolation

The CVS makeup line is isolated following certain events to prevent overfilling of the RCS. In addition, this line is isolated on High 2 containment radioactivity to provide containment isolation following an accident. This line is not isolated on a containment isolation signal, to allow the CVS makeup pumps to perform their defense-in-depth functions. However, if very high containment radioactivity exists (above the High 2 setpoint) this line is isolated.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

A signal to isolate the CVS is derived from two-out-of-four high steam generator levels on either steam generator, two-out-of-four channels of pressurizer level indicating high or two-out-of-four channels of containment radioactivity indicating high. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

16.a. Steam Generator Narrow Range Water Level – High 2

Four channels of steam generator level are provided for each steam generator. Two-out-of-four channels on either steam generator indicating level greater than the setpoint will close the isolation valves for the CVS. This Function prevents adding makeup water to the RCS during a SGTR. This Function is required to be OPERABLE in MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS. This Function is not applicable in MODES 3 and 4 if the CVS makeup flowpath is isolated. This Function is not required to be OPERABLE in MODES 5 and 6 because the RCS pressure and temperature are reduced and a steam generator tube rupture event is not credible.

16.b. Pressurizer Water Level – High 1 Coincident with Safeguards Actuation

Four channels of pressurizer level are provided on the pressurizer. Two-out-of-four channels on indicating level greater than the High 1 setpoint coincident with a Safeguards Actuation signal (Function 1) will close the containment isolation valves for the CVS. This Function prevents the pressurizer level from reaching a level that could lead to water relief through the pressurizer safety valves during some DBAs. This Function is required to be OPERABLE in MODES 1, 2, and 3. This function is not required to be OPERABLE in MODES 4, 5, and 6, because it is not required to mitigate a DBA in these MODES. This Function is not applicable in MODE 3, if the CVS makeup flowpath is isolated.

16.c. Pressurizer Water Level – High 2

A signal to close the CVS isolation valves is generated on Pressurizer Water Level – High 2. This Function results from the coincidence of pressurizer level above the High 2 setpoint in any two of the four divisions. This Function is automatically blocked when the pressurizer pressure is below the P-11 permissive setpoint to permit pressurizer water solid conditions

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

with the plant cold and to permit level makeup during plant cooldowns. This Function is automatically unblocked when RCS pressure is above the P-19 setpoint. This Function is required to be OPERABLE in MODES 1, 2, and 3 and in MODE 4 when the RCS is not being cooled by the RNS. This Function is not required to be OPERABLE in MODE 4 if the CVS makeup flowpath is isolated. This Function is not required to be OPERABLE in MODES 5 and 6 because it is not required to mitigate a DBA in these MODES.

16.d. Containment Radioactivity – High 2

Four channels of Containment Radioactivity – High 2 are required to be OPERABLE in MODES 1, 2, and 3 when the potential exists for a LOCA, to ensure that the radioactivity inside containment is not released to the atmosphere. This Function is not required to be OPERABLE in MODE 3 if the associated flowpath is isolated. This signal results from the coincidence of containment radioactivity above the High 2 Setpoint in any two of the four divisions. These Functions are

not required to be OPERABLE in MODES 4, 5, and 6 because there is no credible release of radioactivity into the containment in these MODES that would result in a High 2 actuation.

16.e. Manual Initiation

Manual Chemical Volume Control System Makeup Isolation is actuated by either of two switches in the main control room. Either switch closes Chemical Volume Control System Makeup valves. The LCO requires two switches to be OPERABLE.

16.f. Source Range Neutron Flux Doubling (Function 15.a)

Chemical Volume Control System Makeup Isolation is actuated by the Source Range Neutron Flux Doubling Function. The Source Range Neutron Flux Doubling Function requirements are the same as the requirements for Boron Dilution Block Function 15.a, Source Range Neutron Flux Doubling. Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 15.a is referenced for all requirements.

17. Normal Residual Heat Removal System Isolation

The RNS suction line is isolated by closing the containment isolation

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

valves on High 2 containment radioactivity to provide containment isolation following an accident. This line is isolated on a safeguards actuation signal. However, the valves may be reset to permit the RNS pumps to perform their defense-in-depth functions post-accident. Should a high containment radiation signal (above the High 2 setpoint) develop following the containment isolation signal, the RNS valves would re-close. A high containment radiation signal is indicative of a high RCS source term and the valves would re-close to assure offsite doses do not exceed regulatory limits.

17.a. <u>Containment Radioactivity – High 2</u>

A signal to isolate the normal residual heat removal system is generated from the coincidence of containment radioactivity above the High 2 setpoint in two-out-of-four channels. Four channels of Containment Radioactivity – High 2 are required to be OPERABLE in MODES 1, 2, and 3 when the potential exists for a LOCA, to ensure that the radioactivity inside containment is not released to the atmosphere. This Function is not required to be OPERABLE in MODE 3 if the RNS suction line is isolated. These Functions are not required to be OPERABLE in MODES 4, 5, and 6 because no DBA that could release radioactivity into the containment is considered credible in these MODES.

17.b. <u>Safeguards Actuation (Function 1)</u>

This Function is also initiated by all Functions that initiated the Safeguards Actuation signal. The requirements to isolate the normal residual heat removal system are the same as the requirements for the Safeguards Actuation Function. Therefore, the requirements are not repeated in Table 3.3.2.1. Instead, Function 1 is referenced for all initiating Functions and requirements.

17.c. <u>Manual Initiation</u>

The operator can initiate RNS isolation at any time from the control room by simultaneously actuating two switches in the same actuation set. Because an inadvertent actuation of RNS isolation could have serious consequences, two switches must be actuated simultaneously to initiate isolation. There are two sets of two switches in the control room. Simultaneously actuating the two switches in either set will isolate the RNS in the same manner as the automatic actuation signal. Two Manual Initiation switches in each set are required to be OPERABLE to ensure no single failure disables the Manual

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Initiation Function.

18. ESFAS Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions backup manual actions to ensure bypassable Functions are in operation under the conditions assumed in the safety analyses.

18.a. Reactor Trip, P-4

There are eight reactor trip breakers with two breakers in each division. The P-4 interlock is enabled when the breakers in two-out-of-four divisions are open. Additionally, the P-4 interlock is enabled by all Automatic Reactor Trip Actuations. The Functions of the P-4 interlock are:

- Trip the main turbine
- Block boron dilution
- Isolate main feedwater coincident with low reactor coolant temperature (This function is not assumed in safety analysis therefore, it is not included in the technical specifications.)

The reactor trip breaker position switches that provide input to the P-4 interlock only function to energize or de-energize or open or close contacts. Therefore, this Function has no adjustable Trip Setpoint.

This Function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality. This Function does not have to be OPERABLE in MODE 4, 5, or 6 to trip the main turbine, because the main turbine is not in operation.

The P-4 Function does not have to be OPERABLE in MODE 4 or 5 to block boron dilution, because Function 15.a, Source Range Neutron Flux Doubling, provides the required block. In MODE 6, the P-4 interlock with the Boron Dilution Block Function is not required, since the unborated water source flow path isolation valves are locked closed in accordance with

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LCO 3.9.2.

18.b. Pressurizer Pressure, P-11

The P-11 interlock permits a normal unit cooldown and depressurization without Safeguards Actuation or main steam line and feedwater isolation. With pressurizer pressure channels less than the P-11 setpoint, the operator can manually block the Pressurizer pressure – Low, Steam Line Pressure – Low, and T_{cold} – Low Safeguards Actuation signals and the Steam Line Pressure – Low and T_{cold} – Low steam line

isolation signals. When the Steam Line Pressure – Low is manually blocked, a main steam isolation signal on Steam Line Pressure-Negative Rate – High is enabled. This provides protection for an SLB by closure of the main steam isolation valves. Manual block of feedwater isolation on T_{avg} – Low 1, Low 2, and T_{cold} – Low is also permitted below P-11. With pressurizer pressure channels ≥ P-11 setpoint, the Pressurizer Pressure – Low, Steam Line Pressure – Low, and T_{cold} – Low Safeguards Actuation signals and the Steam Line Pressure Low and T_{cold} – Low steam line isolation signals are automatically enabled. The feedwater isolation signals on T_{cold} – Low, T_{avg} – Low 1 and Low 2 are also automatically enabled above P-11. The operator can also enable these signals by use of the respective manual reset buttons. When the Steam Line Pressure – Low and T_{cold} – Low steam line isolation signals are enabled, the main steam isolation on Steam Line Pressure-Negative Rate – High is disabled. The Setpoint reflects only steady state instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the Safeguards Actuation or main steam or feedwater isolation. This Function does not have to be OPERABLE in MODE 4, 5, or 6, because plant pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

18.c. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when the respective NIS intermediate range channel goes approximately one decade above the minimum channel reading. Below the setpoint, the P-6 interlock automatically unblocks the flux doubling function, permitting the block of boron dilution. Normally, this Function is blocked by the main

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

control room operator during reactor startup. This Function is required to be OPERABLE in MODE 2.

18.d. Pressurizer Level, P-12

The P-12 interlock is provided to permit midloop operation without core makeup tank actuation, reactor coolant pump trip, CVS letdown isolation, or purification line isolation. With pressurizer level channels less than the P-12 setpoint, the

operator can manually block low pressurizer level signal used for these actuations. Concurrent with blocking CMT actuation on low pressurizer level, ADS 4th Stage actuation on Low 2 RCS hot leg level is enabled. Also CVS letdown isolation on Low 1 RCS hot leg level is enabled. When the pressurizer level is above the P-12 setpoint, the pressurizer level signal is automatically enabled and a confirmatory open signal is issued to the isolation valves on the CMT cold leg balance lines. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

18.e. RCS Pressure, P-19

The P-19 interlock is provided to permit water solid conditions (i.e., when the pressurizer water level is >92%) in lower MODES without automatic isolation of the CVS makeup pumps. With RCS pressure below the P-19 setpoint, the operator can manually block CVS isolation on High 2 pressurizer water level, and block Passive RHR actuation and Pressurizer Heater Trip on High 3 pressurizer water level. When RCS pressure is above the P-19 setpoint, these Functions are automatically unblocked. This Function is required to be OPERABLE IN MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS. When the RNS is cooled by the RNS, the RNS suction relief valve provides the required overpressure protection (LCO 3.4.14).

18.f. Reactor Trip Breaker Open, P-3

The P-3 interlock is provided to permit the block of automatic Safeguards Actuation after a predetermined time interval following automatic Safeguards Actuation.

The reactor trip breaker position switches that provide input to the P-3 interlock only function to energize or de-energize (open or close) contacts. Therefore, this Function does not

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

have an adjustable Trip Setpoint.

19. <u>Containment Air Filtration System Isolation</u>

Some DBAs such as a LOCA may release radioactivity into the containment where the potential would exist for the radioactivity to be released to the atmosphere and exceed the acceptable site dose limits. Isolation of the Containment Air Filtration System provides protection to prevent radioactivity inside containment from being released to the atmosphere.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

19.a. Containment Radioactivity – High 1

Three channels of Containment Radioactivity – High 1 are required to be OPERABLE in MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS, when the potential exists for a LOCA, to protect against radioactivity inside containment being released to the atmosphere. These Functions are not required to be OPERABLE in MODE 4 with the RCS being cooled by the RNS or MODES 5 and 6, because any DBA release of radioactivity into the containment in these MODES would not require containment isolation.

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19.b. Containment Isolation (Function 3)

Containment Air Filtration System Isolation is also initiated by all Functions that initiate Containment Isolation. The Containment Air Filtration System Isolation requirements for these Functions are the same as the requirements for the Containment Isolation. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 3, Containment Isolation, is referenced for initiating Functions and requirements.

20. Main Control Room Isolation and Air Supply Initiation

Isolation of the main control room and initiation of the air supply provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. This Function is required to be OPERABLE in MODES 1, 2, 3, and 4, and during movement of irradiated fuel because of the potential for a fission product release following a fuel handling accident, or other DBA.

20.a. Control Room Air Supply Radiation – High 2

Two radiation monitors are provided on the main control room air intake. If either monitor exceeds the High 2 setpoint, control room isolation is actuated.

20.b. Pressurizer Pressure – Low

This signal provides protection against a potential release of radioactivity due to a LOCA. The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment). Therefore, the Trip Setpoint reflects the inclusion of both steady-state and adverse environmental instrument uncertainties.

The LCO requires four channels of Pressurizer Pressure – Low to be OPERABLE in MODES 1, 2, and 3 (above P-11, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F), to mitigate the consequences of a high energy line rupture inside containment. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This signal may be manually blocked by the operator below the P-11

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setpoint.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint.

21. Auxiliary Spray and Purification Line Isolation

The CVS maintains the RCS fluid purity and activity level within acceptable limits. The CVS purification line receives flow from the discharge of the RCPs. The CVS also provides auxiliary spray to the pressurizer. To preserve the reactor coolant pressure in the event of a break in the CVS loop piping, the purification line and the auxiliary spray line are isolated on a pressurizer water level Low 1 setpoint. This helps maintain reactor coolant system inventory.

21.a. Pressurizer Water Level – Low 1

A signal to isolate the purification line and the auxiliary spray line is generated upon the coincidence of pressurizer level below the Low 1 setpoint in any two-out-of-four divisions. This Function is required to be OPERABLE in MODES 1 and 2 to help maintain RCS inventory. In MODES 3, 4, 5, and 6, this Function is not needed for accident detection and mitigation.

21.b. Manual Chemical Volume Control System Makeup Isolation (Function 16.e)

The Auxiliary Spray and Purification Line Isolation is also initiated by the Manual Chemical Volume Control System Makeup Isolation Function. The requirements for this Function are the same as the requirements for Manual Chemical Volume Control System Makeup Isolation (Function 16.e), but only apply in MODES 1 and 2. Therefore, the requirements are not repeated in Table 3.3.2-1, and Function 16.e is referenced for all requirements.

22. IRWST Injection Line Valve Actuation

The PXS provides core cooling by gravity injection and recirculation for decay heat removal following an accident. The IRWST has two injection flow paths. Each injection path includes a normally open motor operated isolation valve and two parallel lines, each isolated by one check valve and one squib valve in series. Manual initiation or automatic actuation on an ADS Stage 4 actuation signal or a coincident RCS Loops 1 and 2 Hot Leg Level-Low will generate a signal to open the IRWST injection line and actuate IRWST injection.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

22.a. Manual Initiation

The operator can open IRWST injection line valves at any time from the main control room by actuating two IRWST injection actuation switches in the same actuation set. There are two sets of two switches each in the main control room. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

22.b. ADS Stage 4 Actuation (Function 10)

An open signal will be issued to the IRWST injection isolation valves when an actuation signal is issued to the ADS Stage 4 valves. The requirements for this function are the same as the requirements for the ADS Stage 4 Actuation Function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 10 is referenced for all initiating functions and requirements.

23. IRWST Containment Recirculation Valve Actuation

The PXS provides core cooling by gravity injection and recirculation for decay heat removal following an accident. The PXS has two containment recirculation flow paths. Each path contains two parallel flow paths, one path is isolated by a motor operated valve in series with a squib valve and one path is isolated by a check valve in series with a squib valve. Manual initiation or automatic actuation on a Safeguards Actuation signal coincident with a Low 3 level signal in the IRWST will open these valves.

23.a. Manual Initiation

The operator can open the containment recirculation valves at any time from the main control room by actuating two containment recirculation actuation switches in the same actuation set. There are two sets of two switches each in the main control room. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

23.b. ADS Stage 4 Actuation Coincident with IRWST Level – Low 3

A low IRWST level coincident with a ADS Stage 4 Actuation signal will open the containment recirculation valves. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this trip Function. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6, except when the ADS Stage 4 valves are open or an equivalent relief area is open. The required ADS valves or equivalent relief area are specified in LCO 3.4.12, ADS – Shutdown, RCS Intact and LCO 3.4.13, ADS – Shutdown, RCS Open.

24. Refueling Cavity Isolation

The containment isolation valves in the lines between the refueling cavity and the Spent Fuel Pool Cooling System are isolated on a Low spent fuel pool level.

24.a. Spent Fuel Pool Level – Low

In the event of a leak in the non-safety Spent Fuel Pool Cooling System, closure of the containment isolation valves on low spent fuel pool level in two of three channels will terminate draining of the refueling cavity. Since the transfer canal is open in MODE 6, the spent fuel pool level is the same as the refueling cavity.

Draining of the spent fuel pool, directly, through a leaking Spent Fuel Pool Cooling System is limited by the location of the suction piping, which is near the top of the pool. Therefore, closure of the containment isolation valves between the refueling cavity and the Spent Fuel Pool Cooling System is sufficient to terminate refueling cavity and spent fuel pool leakage through the Spent Fuel Pool Cooling System. This Function is required in MODE 6 to maintain water inventory in the refueling cavity.

25. ESF Logic

This LCO requires four sets of ESF coincidence logic, each set with one battery backed logic group OPERABLE to support automatic actuation. These logic groups are implemented as processor based actuation subsystems. The ESF coincidence logic provides the system level logic interfaces for the divisions.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

25.a. Coincidence Logic

If one division of battery backed coincidence logic is OPERABLE, an additional single failure will not prevent ESF actuations because three divisions will still be available to provide redundant actuation for all ESF Functions. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

26. ESF Actuation

This LCO requires that for each division of ESF actuation, one battery backed logic group be OPERABLE to support both automatic and manual actuation. The ESF actuation subsystems provide the logic and power interfaces for the actuated components.

26.a. Actuation Subsystem

If one battery backed logic group is OPERABLE for the ESF actuation subsystem in all four divisions, an additional single failure will not prevent ESF actuations because ESF actuation subsystems in the other three divisions are still available to provide redundant actuation for ESF Functions. The remaining cabinets in the division with a failed ESF actuation cabinet are still OPERABLE and will provide their ESF Functions. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

The ESFAS instrumentation satisfies Criterion 3 of the 10 CFR 50.36(c)(2)(ii).

27. Pressurizer Heater Trip

Pressurizer heaters are automatically tripped upon receipt of a core makeup tank operation signal or a Pressurizer Water Level – High 3 signal. This pressurizer heater trip reduces the potential for steam generator overfill and automatic ADS Stages 1, 2, and 3 actuation for a steam generator tube rupture event. Automatically tripping the pressurizer heaters reduces the pressurizer level swell for certain non-LOCA events such as loss of normal feedwater, inadvertent CMT operation, and CVS malfunction resulting in an increase in RCS inventory. For small break LOCA analysis, tripping the pressurizer heaters supports depressurization of the RCS following actuation of the CMTs.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

27.a. CMT Actuation (Function 2)

A signal to trip the pressurizer heaters is generated on a CMT actuation signal. The requirements for this function are the same as the requirements for the CMT Actuation Function, except this function is only required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the RCS is not being cooled by the RNS and above the P-19 (RCS pressure) interlock. Therefore, the requirements are not repeated in Table 3.3.2.1. Instead, Function 2 is referenced for initiating Functions and requirements and SR 3.3.2.9 also applies.

27.b. Pressurizer Water Level – High 3

A signal to trip the pressurizer heaters is generated when the pressurizer water level reaches its High 3 setpoint. This signal provides protection against a pressurizer overfill following an inadvertent core makeup tank actuation with consequential loss of offsite power. This Function is automatically unblocked when RCS pressure is above the P-19 setpoint. This Function is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the RCS is not being cooled by the RNS and above the P-19 (RCS pressure) interlock. This Function is not required to be OPERABLE in MODES 5 and 6 because it is not required to mitigate DBA in these MODES.

28. Chemical and Volume Control System Letdown Isolation

The CVS provides letdown to the liquid radwaste system to maintain the pressurizer level. To help maintain RCS inventory in the event of a LOCA, the CVS letdown line is isolated on a Low 1 hot leg level signal in either of the RCS hot leg loops. This Function is required to be OPERABLE in MODE 4 with the RCS being cooled by the RNS. This Function is also required to be OPERABLE in MODE 5, and in MODE 6 with the water level < 23 feet above the top of the reactor vessel flange.

28.a. Hot Leg Level – Low 1

A signal to isolate the CVS letdown valves is generated upon the occurrence of a Low 1 hot leg level in either of the two RCS hot leg loops. This helps to maintain reactor system inventory in the event of a LOCA. This function can be blocked in Modes 1, 2 and 3 and is automatically reset when

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

P-12 is first activated. This function may be manually reset as well. These letdown valves are also closed by all of the initiating Functions and requirements that generate the Containment Isolation Function in Function 3.

29. SG Power Operated Relief Valve and Block Valve Isolation

The Function of the SG Power Operated Relief Valve and Block Valve Isolation is to ensure that the SG PORV flow paths can be isolated during a SG tube rupture (SGTR) event. The PORV flow paths must be isolated following a SGTR to minimize radiological releases from the ruptured steam generator into the atmosphere. The PORV flow path is assumed to open due to high secondary side pressure, during the SGTR. Dose analyses take credit for subsequent isolation of the PORV flow path by the PORV and/or the block valve which receive a close signal on low steam line pressure. Additionally, the PORV flow path can be isolated manually.

This Function is required to be OPERABLE in MODES 1, 2, 3, and 4 with the RCS cooling not being provided by the Normal Residual Heat Removal System (RNS). In MODE 4 with the RCS cooling being provided by the RNS and in MODES 5 and 6, the steam generators are not being used for RCS cooling and the potential for a SGTR is minimized due to the reduced mass and energy in the RCS and steam generators.

29.a. Manual Initiation

Manual initiation of SG Power Operated Relief Valve and Block Valve Isolation can be accomplished from the control room. There are two switches in the control room and either switch can close the SG PORVs and PORV block valves. The LCO requires two switches to be OPERABLE.

29.b. Steam Line Pressure – Low

Steam Line Pressure – Low provides closure of the PORV flow paths in the event of SGTR in which the PORV(s) open, to limit the radiological releases from the ruptured steam generator into the atmosphere.

This Function is anticipatory in nature and has a typical leading/lag ratio of 50/5.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The LCO requires four channels of Steam Line Pressure – Low Function to be OPERABLE in MODES 1, 2, 3, and 4 with the RCS cooling not being provided by the RNS. Four channels are provided in each steam line to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this Function.

ESFAS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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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 on Table 3.3.2-1. The Completion Time(s) of the inoperable equipment of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A second Note has been added to provide clarification that, more than one Condition is listed for each of the Functions in Table 3.3.2-1. If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the second Condition shall be entered.

In the event a channel's Nominal Trip Setpoint is not met, or the transmitter, or the Protection and Safety Monitoring System Division, associated with a specific Function is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the particular protection Function(s) affected. When the Required Channels are specified only on a per steam line, per loop, per SG, basis, then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 in MODES 1 through 4 and LCO 3.0.8 for MODE 5 and 6 should be immediately entered if applicable in the current MODE of operation.

<u>A.1</u>

Condition A is applicable to all ESFAS protection Functions. Condition A addresses the situation where one or more channels/divisions for one or more functions are inoperable at the same time. The Required Action is

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to refer to Table 3.3.2-1 and to take the Required Actions for the protection Functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1 and B.2

With one or two channels or divisions inoperable, one affected channel or division must be placed in a bypass or trip condition within 6 hours. If one channel or division is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels or divisions will not prevent the protective function.) If one channel or division is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels or divisions will not prevent the protective function.) If one channel or division is bypassed and one channel or division is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channel(s) or division(s) in the bypassed or tripped condition is justified in Reference 6.

C.1

With one channel inoperable, the affected channel must be placed in a bypass condition within 6 hours. The 6 hours allowed to place the inoperable channel in the bypass condition is justified in Reference 6. If one CVS isolation channel is bypassed, the logic becomes one-out-of-one. A single failure in the remaining channel could cause a spurious CVS isolation. Spurious CVS isolation, while undesirable, would not cause an upset plant condition.

D.1

With one required division inoperable, the affected division must be restored to OPERABLE status within 6 hours.

Condition D applies to one inoperable required division of the P-3 & P-4 Interlocks (Functions 18.a and 18.f). With one required division inoperable, the 2 remaining OPERABLE divisions are capable of providing the required interlock function, but without a single failure. The P-3 & P-4 Interlocks are enabled when RTBs in two divisions are detected as open. The status of the other inoperable, non-required P-3 & P-4 division is not significant, since P-3 & P-4 divisions can not be tripped or bypassed. In order to provide single failure tolerance, 3 required divisions must be OPERABLE.

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Condition D also applies to one inoperable division of ESF coincidence logic or ESF actuation (Functions 25 and 26). The ESF coincidence logic and ESF actuation divisions are inoperable when their associated battery-backed subsystem is inoperable. With one inoperable division, the 3 remaining OPERABLE divisions are capable of mitigating all DBAs, but without a single failure.

The 6 hours allowed to restore the inoperable division is reasonable based on the capability of the remaining OPERABLE divisions to mitigate all DBAs and the low probability of an event occurring during this interval.

E.1

Condition E is applicable to manual initiation of:

- Safeguards Actuation;
- CMT Actuation;
- Containment Isolation;
- Steam Line Isolation;
- Main Feedwater Control Valve Isolation;
- Main Feedwater Pump Trip and Valve Isolation;
- ADS Stages 1, 2, & 3 Actuation;
- ADS Stage 4 Actuation;
- Passive Containment Cooling Actuation;
- PRHR Heat Exchanger Actuation;
- CVS Makeup Line Isolation;
- IRWST Injection Line Valve Actuation;
- IRWST Containment Recirculation Valve Actuation;
- Steam Generator PORV Flow Path Isolation

This Action addresses the inoperability of the system level manual initiation capability for the ESF Functions listed above. With one switch or switch set inoperable for one or more Functions, the system level manual initiation capability is reduced below that required to meet single

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failure criterion. Required Action E.1 requires the switch or switch set for

system level manual initiation to be restored to OPERABLE status within 48 hours. The specified Completion Time is reasonable considering that the remaining switch or switch set is capable of performing the safety function.

<u>F.1, F.2.1, and F.2.2</u>

Condition F is applicable to the Main Control Room (MCR) isolation and air supply initiation function which has only two channels of the initiating process variable. With one channel inoperable, the logic becomes one-out-of-one and is unable to meet single failure criterion. Restoring all channels to OPERABLE status ensures that a single failure will not prevent the protective Function.

Alternatively, radiation monitor(s) which provide equivalent information and control room isolation and air supply initiation manual controls may be verified to be OPERABLE. These provisions for operator action can replace one channel of radiation detection and system actuation. The 72 hour Completion Time is reasonable considering that there is one remaining channel OPERABLE and the low probability of an event occurring during this interval.

<u>G.1</u>

With one switch, switch set, channel, or division inoperable, the system level initiation capability is reduced below that required to meet single failure criterion. Therefore, the required switch, switch set, channel, and division must be returned to OPERABLE status within 72 hours. The specified Completion Time is reasonable considering the remaining switch, switch set, channel, or division is capable of performing manual initiation.

<u>H.1</u>

With one channel inoperable, the inoperable channel must be placed in a trip condition within 6 hours.

Condition H is applicable to the PRHR heat exchangers actuation on SG Narrow Range Water Level Low coincident with Startup Feedwater Flow Low (Function 13.b). With one startup feedwater channel inoperable, the inoperable channel must be placed in a trip condition within 6 hours. If one channel is tripped, the interlock condition is satisfied. Condition H is also applicable to Refueling Cavity Isolation (Function 24.a). With one of the three spent fuel pool level channels

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inoperable, the inoperable channel must be placed in a trip condition within 6 hours. If one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The specified Completion Time is reasonable considering the time required to complete this action.

I.1 and I.2

Condition I applies to IRWST containment recirculation valve actuation on safeguards actuation coincident with IRWST Level Low 3 (Function 23.b). With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within 6 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 6.

J.1 and J.2

Condition J applies to the P-6, P-11, P-12, and P-19 interlocks. With one or two required channel(s) inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour, or any Function channels associated with inoperable interlocks placed in a bypassed condition within 7 hours. Verifying the interlock state manually accomplishes the interlock role.

If one interlock channel is inoperable, the associated Function(s) must be placed in a bypass or trip condition within 7 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.)

If two interlock channels are inoperable, one channel of the associated Function(s) must be bypassed and one channel of the associated

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Function(s) must be tripped. In this state, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 7 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 6.

K.1

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

Condition K is applicable to the MCR Isolation and Air Supply Initiation (Function 20), during movement of irradiated fuel assemblies. If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must suspend movement of the irradiated fuel assemblies immediately. The required action suspends activities with potential for releasing radioactivity that might enter the MCR. This action does not preclude the movement of fuel to a safe position.

L.1

If the required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a MODE in which the LCO does not apply. This accomplished by placing the plant in MODE 3 within 6 hours. The allowed time is reasonable, based operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

M.1 and M.2

If the Required Action and associated Completion Time of the first condition listed in Table 3.3.2-1 is not met, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant 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 plant conditions from full power conditions in an orderly manner without challenging plant systems.

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N.1 and N.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 with the RCS being cooled by the RNS 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 without challenging plant systems.

O.1 and O.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in 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 plant conditions from full power conditions in an orderly manner without challenging plant systems.

P.1, P.2.1, and P.2.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 cannot be met, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer needed. This is accomplished by isolating the affected flow path(s) within 24 hours. By isolating the flow path from the demineralized water storage tank to the RCS, the need for automatic isolation is eliminated.

To assure that the flow path remains closed, the flow path shall be isolated by the use of one of the specified means (P.2.1) or the flow path shall be verified to be isolated (P.2.2). A means of isolating the affected flow path(s) includes at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured within 7 days. If one of the P.2.1 specified isolation means is not used, the affected flow path shall be verified to be isolated once per 7 days.

This action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the

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valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

Q.1, Q.2.1, and Q.2.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer needed. This is accomplished by isolating the affected flow path by the use of at least one closed manual or closed and deactivated automatic valve within 6 hours.

If the flow path is not isolated within 6 hours the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 12 hours and in MODE 4 within 18 hours.

This action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. 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 flow path can be rapidly isolated when a need for flow path isolation is indicated.

R.1, R.2.1.1, R.2.1.2, and R.2.2

If the Required Action and associated Completion Time of the first Condition given in Table 3.3.2-1 is not met the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 3 within 6 hours and isolating the affected flow path(s) within 12 hours. To assure that the flow path remains closed, the affected flow path shall be verified to be isolated once per 7 days.

If the flow path is not isolated within 12 hours the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 4 with the RCS cooling provided by the RNS within 30 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

This action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the

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valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

S.1, S.2.1.1, S.2.1.2, S.2.1.3, and S.2.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 with the RCS cooling provided by the RNS within 24 hours. Once the plant has been placed in MODE 4 the affected flow path must be isolated within 30 hours. To assure that the flow path remains closed, the affected flow path shall be verified to be isolated once per 7 days.

If the flow path is not isolated within 12 hours, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 5 within 42 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

This action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. 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 flow path can be rapidly isolated when a need for flow path isolation is indicated.

T.1.1, T.1.2.1, T.1.2.2, T.2.1, and T.2.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a Condition in which the likelihood and consequences of an event are minimized. This is accomplished by isolating the affected flow path within 6 hours and isolating the affected flow path(s) by the use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured within 7 days or verify the affected flow path is isolated once per 7 days.

If the flow path is not isolated within 6 hours the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 12 hours and in MODE 5 within 42 hours.

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The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

This action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. 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 flow path can be rapidly isolated when a need for flow path isolation is indicated.

U.1 and U.2

If the Required Action and the associated Completion Time of the first Condition given in Table 3.3.2-1 is not met, and the required switch or switch set is not restored to OPERABLE status within 48 hours, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 5 within 12 hours. Once in MODE 5, action shall be immediately initiated to open the RCS pressure boundary and establish ≥ 20% pressurizer level. The 12 hour Completion Time is a reasonable time to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems. Opening the RCS pressure boundary assures that cooling water can be injected without ADS operation. Filling the RCS to provide ≥ 20% pressurizer level minimizes the consequences of a loss of decay heat removal event.

V.1, V.2.1, and V.2.2

If the Required Action and the associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met and the required channel(s) is not bypassed within 6 hours, the inoperable channel(s) must be restored within 168 hours. The 168 hour Completion Time is based on the ability of the two remaining OPERABLE channels to provide the protective Function even with a single failure.

If the channel(s) is not restored within the 168 hour Completion Time, the plant shall be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 5 within 180 hours (the next 12 hours). Once in MODE 5, action shall be initiated to open the RCS pressure boundary and establish ≥ 20% pressurizer level. The 12 hours is a reasonable time

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to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems.

Opening the RCS pressure boundary assures that cooling water can be injected without ADS operation. Filling the RCS to provide ≥ 20% pressurizer level minimizes the consequences of a loss of decay heat removal event.

<u>W.1, W.2, W.3, and W.4</u>

If the Required Action and the associated Completion Time listed in Table 3.3.2-1 is not met while in MODES 5 and 6, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. This is accomplished by immediately initiating action to be in MODE 5 with the RCS open and ≥ 20% pressurizer level or to be in MODE 6 with the upper internals removed. The flow path from the demineralized water storage tank to the RCS shall also be isolated by the used of at least one closed and de-activated automatic valve or closed manual valve. These requirements minimize the consequences of the loss of decay heat removal by maximizing RCS inventory and maintaining RCS temperature as low as practical. Additionally, the potential for a criticality event is minimized by isolation of the demineralized water storage tank and by suspension of positive reactivity additions.

<u>X.1, X.2, and X.3</u>

If the Required Action and the associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met while in MODES 5 and 6, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. This is accomplished by immediately initiating action to be in MODE 5 with the RCS open and ≥ 20% pressurizer level or to be in MODE 6 with the upper internals removed. These requirements minimize the consequences of the loss of decay heat removal by maximizing RCS inventory and maintaining RCS temperature as low as practical. Additionally, the potential for a criticality event is minimized by suspension of positive reactivity additions.

<u>Y.1, Y.2, Y.3, and Y.4</u>

If the Required Action and the associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met while in MODE 4, with RCS cooling provided by the RNS, MODE 5, or MODE 6, the plant must be placed in a MODE in which the likelihood and consequences of an event

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are minimized. If in MODE 4, this is accomplished by placing the plant in MODE 5 within 12 hours. The 12 hours is a reasonable time to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems.

If in MODE 4 or 5, Required Action Y.3 requires initiation of action within 12 hours to close the RCS pressure boundary and establish ≥ 20% pressurizer level. The 12 hour Completion Time allows transition to MODE 5 in accordance with Y.2, if needed, prior to initiating action to open the RCS pressure boundary.

If in MODE 6, Required Action Y.4 requires the plant to be maintained in MODE 6 and initiation of action to establish the reactor cavity water level ≥ 23 feet above the top of the reactor vessel flange.

Required Actions Y.2, Y.3, and Y.4 minimize the consequences of a loss of decay heat removal event by optimizing conditions for RCS cooling in MODE 5 using the PRHR HX or in MODE 6 using IRWST injection. Additionally, maximizing RCS inventory and maintaining RCS temperature as low as practical further minimize the consequences of a loss of decay heat removal event. Closing the RCS pressure boundary in MODE 5 assures that PRHR HX cooling is available. Additionally, the potential for a criticality event is minimized by suspension of positive reactivity additions.

<u>Z.1, Z.2.1, and Z.2.2</u>

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer needed. This is accomplished by isolating the affected flow path by the use of at least one closed manual or closed and deactivated automatic valve within 6 hours.

If the flow path is not isolated within 6 hours, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 12 hours and in MODE 4 with RCS cooling provided by the RNS within 30 hours.

This Action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the

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valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

AA.1.1, AA.1.2.1, AA1.2.2, AA.2.1, AA.2.2, and AA.2.3

If the Required Action and associated Completion Time of the first condition listed in Table 3.3.2-1 is not met, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer needed. This is accomplished by isolating the affected flow path within 24 hours. By isolating the CVS letdown flow path from the RCS, the need for automatic isolation is eliminated.

To assure that the flow path remains closed, the flow path shall be isolated by the use of one of the specified means (AA.1.2.1) or the flow path shall be verified to be isolated (AA.1.2.2). A means of isolating the affected flow path includes at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured, within 7 days. If one of the P.2.1 specified isolation means is not used, the affected flow path shall be verified to be isolated once per 7 days.

This action is modified by a Note allowing the flow path to be unisolated intermittently under administrative control. 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 flow path can be rapidly isolated when a need for flow path isolation is indicated.

If the flow path cannot be isolated in accordance with Required Actions AA.1.1, AA.1.2.1 and AA.1.2.2, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. If in MODE 4, this is accomplished by placing the plant in MODE 5 within 12 hours. The 12 hours is a reasonable time to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems.

If in MODE 4 or 5, Required Action AA.2.2 requires initiation of action, within 12 hours, to establish > 20% pressurizer level. The 12 hour Completion Time allows transition to MODE 5 in accordance with AA.2.1, if needed, prior to initiating action to establish the pressurizer level.

ACTIONS (continued)

If in MODE 6, Required Action AA.2.3 requires the plant to be maintained in MODE 6 and initiation of action to establish the reactor cavity water level ≥ 23 feet above the top of the reactor vessel flange.

Required Actions AA.2.2 and AA.2.3 minimize the consequences of an event by optimizing conditions for RCS cooling in MODE 5 using the PRHR HX or in MODE 6 using IRWST injection.

BB.1 and BB.2

With one channel inoperable, the inoperable channel must be placed in bypass and the hot leg level continuously monitored.

If one channel is placed in bypass, automatic actuation will not occur. Continuous monitoring of the hot leg level provides sufficient information to permit timely operator action to ensure that ADS Stage 4 actuation can occur, if needed to mitigate events requiring RCS makeup, boration, or core cooling. Operator action to manually initiate ADS Stage 4 actuation is assumed in the analysis of shutdown events (Reference 11). It is also credited in the shutdown PRA (Reference 12) when automatic actuation is not available.

SURVEILLANCE REQUIREMENTS

The Surveillance Requirements for each ESF Function are identified by the Surveillance Requirements column of Table 3.3.2-1. A Note has been added to the Surveillance Requirement table to clarify that Table 3.3.2-1 determines which Surveillance Requirements apply to which ESF Functions.

SR 3.3.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is based on operating experience that demonstrates channel failure is rare. Automated operator aids may be used to facilitate performance of the CHANNEL CHECK.

<u>SR 3.3.2.2</u>

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. This test, in conjunction with the ACTUATION DEVICE TEST, demonstrates that the actuated device responds to a simulated actuation signal. The ESF coincidence logic and ESF actuation subsystems within a division are tested every 92 days on a STAGGERED TEST BASIS.

A test subsystem is provided with the protection and safety monitoring system to aid the plant staff in performing the ACTUATION LOGIC TEST. The test subsystem is designed to allow for complete functional testing by using a combination of system self-checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded.

For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Generally this verification includes a comparison of the outputs from two or more redundant subsystems or channels.

Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing.

To the extent possible, protection and safety monitoring system functional testing is accomplished with continuous system self-checking features and the continuous functional testing features. The ACTUATION LOGIC TEST shall include a review of the operation of the test subsystem to verify the completeness and adequacy of the results.

If the ACTUATION LOGIC TEST can not be completed using the built-in test subsystem, either because of failures in the test subsystem or failures in redundant channel hardware used for functional testing, the ACTUATION LOGIC TEST can be performed using portable test equipment.

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of every 92 days on a STAGGERED TEST BASIS provides a complete test of all four divisions once per year. This frequency is adequate based on the inherent high reliability of the solid state devices which comprise this equipment; the additional reliability provided by the redundant subsystems; and the use of continuous diagnostic test features, such as deadman timers, memory checks, numeric coprocessor checks, cross-check of redundant subsystems, and tests of timers, counters, and crystal time basis, which will report a failure within these cabinets to the operator.

SR 3.3.2.3

SR 3.3.2.3 is the performance of a TADOT of the manual actuations, initiations, and blocks for various ESF Functions, the reactor trip breaker open (P-3), and the reactor trip (P-4) input from the IPCs. This TADOT is performed every 24 months.

The Frequency is based on the known reliability of the ESF Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The functions have no setpoints associated with them.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a CHANNEL CALIBRATION every 24 months or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor and the IPC.

The Frequency is based on operating experience and consistency with the refueling cycle.

This Surveillance Requirement is modified by a Note. The Note states that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.2.5

SR 3.3.2.5 is the performance of a CHANNEL OPERATIONAL TEST (COT) every 92 days.

SURVEILLANCE REQUIREMENTS (continued)

A COT is performed on each required channel to provide reasonable assurance that the entire channel will perform the intended ESF Function.

A test subsystem is provided with the protection and safety monitoring system to aid the plant staff in performing the COT. The test subsystem is designed to allow for complete functional testing by using a combination of system self-checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded.

For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Generally this verification includes a comparison of the outputs from two or more redundant subsystems or channels.

Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing.

To the extent possible, protection and safety monitoring system functional testing is accomplished with continuous system self-checking features and the continuous functional testing features. The COT shall include a review of the operation of the test subsystem to verify the completeness and adequacy of the results.

If the COT can not be completed using the built-in test subsystem, either because of failures in the test subsystem or failures in redundant channel hardware used for functional testing, the COT can be performed using portable test equipment.

The 92 day Frequency is based on Reference 6 and the use of continuous diagnostic test features, such as deadman timers, cross-check of redundant channels, memory checks, numeric coprocessor checks, and tests of timers, counters and crystal time bases, which will report a failure within the integrated protection cabinets to the operator.

During the COT, the protection and safety monitoring system cabinets in the division under test may be placed in bypass.

SR 3.3.2.6

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis.

SURVEILLANCE REQUIREMENTS (continued)

Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment reaches the required functional state (e.g., valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate Chapter 7 (Ref. 2) response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements” (Ref. 11), provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

ESF RESPONSE TIME tests are conducted on an 24 month STAGGERED TEST BASIS. Testing of the devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 24 months. The 24 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.7

SR 3.3.2.7 is the performance of an ACTUATION DEVICE TEST. This test, in conjunction with the ACTUATION LOGIC TEST, demonstrates that the actuated device responds to a simulated actuation signal. This Surveillance Requirement is applicable to the equipment which is actuated by the Protection Logic Cabinets except squib valves. The OPERABILITY of the actuated equipment is checked by exercising the equipment on an individual basis.

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

This Surveillance Requirement is modified by a Note that states that actuated equipment, that is included in the Inservice Test (IST) Program, is exempt from this surveillance. The IST Program provides for exercising of the safety related valves on a more frequent basis. The results from the IST Program can therefore be used to verify OPERABILITY of the final actuated equipment.

SR 3.3.2.8

SR 3.3.2.8 is the performance of an ACTUATION DEVICE TEST, similar to that performed in SR 3.3.2.7, except this Surveillance Requirement is specifically applicable to squib valves. This test, in conjunction with the ACTUATION LOGIC TEST, demonstrates that the actuated device responds to a simulated actuation signal. The OPERABILITY of the squib valves is checked by performing a continuity check of the circuit from the Protection Logic Cabinets to the squib valve.

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any additional risks associated with inadvertent operation of the squib valves.

SR 3.3.2.9

SR 3.3.2.9 is the performance of an ACTUATION DEVICE TEST. This test, in conjunction with the ACTUATION LOGIC TEST, demonstrates that the actuated device responds to a simulated actuation signal. This Surveillance Requirement is applicable to the circuit breakers which de-energize the power to the pressurizer heaters upon a pressurizer heater trip. The OPERABILITY of these breakers is checked by opening these breakers using the Plant Control System.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation. This Frequency is adequate based on the use of multiple circuit breakers to prevent the failure of any single circuit breaker from disabling the function and that all circuit breakers are tested.

REFERENCES

1. Chapter 6, "Engineered Safety Features."
2. Chapter 7, "Instrumentation and Controls."
3. Chapter 15, "Accident Analysis."
4. Institute of Electrical and Electronic Engineers, IEEE-603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," June 27, 1991.
5. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."
6. APP-GW-GSC-020, "Technical Specification Completion Time and Surveillance Frequency Justification."
7. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
8. NUREG-1218, "Regulatory Analysis for Resolution of USI A-47," 4/88.
9. WCAP-16361-P, "Westinghouse Setpoint Methodology for Protection Systems – AP1000," May 2006 (proprietary).
10. ESBU-TB-97-01, Westinghouse Technical Bulletin, "Digital Process Rack Operability Determination Criteria," May 1, 1997.
11. WCAP-13632-P-A (Proprietary) and WCAP-13787-A (Non-Proprietary), Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.

B 3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM Instrumentation is to display unit variables that provide information required by the main control room operators during accident situations. These plant variables provide the necessary information to assess the process of accomplishing or maintaining critical safety functions. The instruments which monitor these variables are designated in accordance with Reference 1.

The OPERABILITY of the PAM Instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. This capability is consistent with the recommendations of Reference 1.

A PAM CHANNEL shall extend from the sensor up to the display device, and shall include the sensor (or sensors), the signal conditioning, any associated datalinks, the display device, any signal gathering or processing subsystems, and any data processing subsystems. Note that for digital PAM CHANNELs, the information may be displayed on multiple display devices. For this case, the PAM CHANNEL shall extend to any available qualified display device.

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category 1 variables. The unit specific implementation of Regulatory Guide 1.97 has not identified any Type A variables, therefore, only Category 1 variables are specified.

APPLICABLE SAFETY ANALYSES

The PAM Instrumentation ensures that the main control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;
- 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.

APPLICABLE SAFETY ANALYSES (continued)

PAM Instrumentation that is required in accordance with Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for those monitors which provide information required by the control room operators to assess the process of accomplishing or maintaining critical safety functions. This LCO addresses those Regulatory Guide 1.97 instruments which are listed in Table 3.3.3-1.

The OPERABILITY of the PAM Instrumentation ensures there is sufficient information available on selected plant parameters to monitor and assess plant status following an accident. This capability is consistent with the recommendations of Reference 1.

Category 1 non-type A variables are required to meet Regulatory Guide 1.97 Category 1 (Ref. 1) 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 3.3.3-1. Each of these is a Category 1 variable.

1. <u>Intermediate Range Neutron Flux</u>

   Neutron Flux indication is provided to verify reactor shutdown. The neutron flux intermediate range is sufficient to cover the full range of flux that may occur post accident.

   Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

2, 3. <u>Reactor Coolant System (RCS) Wide Range Hot and Cold Leg Temperature</u>

   RCS Hot and Cold Leg Temperatures are provided for verification of core cooling and long-term surveillance. The channels provide indication over a range of 50°F to 700°F.

   In addition to this, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the plant conditions necessary to establish natural circulation in the RCS.

LCO (continued)

4. RCS Pressure

RCS wide range pressure is provided for verification of core cooling and RCS integrity long term surveillance.

5. Pressurizer Pressure and RCS Subcooling Monitor

Pressurizer Pressure is used to determine RCS Subcooling. The RCS Subcooling Monitor is provided for verification of core cooling. Subcooling margin is available when the RCS pressure is greater than the saturation pressure corresponding to the core exit temperature. Inputs to the Subcooling Monitor are pressurizer pressure and RCS hot leg temperature.

6. Containment Water Level

Containment Water Level is used to monitor the containment environment during accident conditions. The containment water level can also provide information to the operators that the various stages of safety injection along with system depressurization are progressing.

7. Containment Pressure

The containment pressure transmitters monitor the containment pressure over the range of -5 to 10 psig. This provides information on post accident containment pressure and containment integrity.

8. Containment Pressure (Extended Range)

The extended range containment pressure transmitters are instruments that operators use for monitoring the potential for breach of containment, a fission product barrier. The extended range sensors monitor containment pressure over the range of 0 to 240 psig.

9. Containment Area Radiation (High Range)

Containment Area Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

LCO (continued)

10. Pressurizer Level and Associated Reference Leg Temperature

Pressurizer level is provided to monitor the RCS coolant inventory. During an accident, operation of the safeguards systems can be verified based on coolant inventory indicators.

The reference leg temperature is included in the Technical Specification since it is used to compensate the level signal.

11. In-Containment Refueling Water Storage Tank (IRWST) Water Level

The IRWST provides a long term heat sink for non-LOCA events and is a source of injection flow for LOCA events. When the IRWST is a heat sink, the level will change due to increased volume associated with the temperature increase. When saturation temperature is reached, the IRWST will begin steaming and initially lose mass to the containment atmosphere until condensation occurs on the steel containment shell which is cooled by the passive containment cooling system. The condensate is returned to the IRWST via a gutter.

During a LOCA, the IRWST is available for injection. Depending on the severity of the event, when a fully depressurized RCS has been achieved, the IRWST will inject by gravity flow.

12. Passive Residual Heat Removal (PRHR) Flow and PRHR Outlet Temperature

PRHR Flow is provided to monitor primary system heat removal during accident conditions when the steam generators are not available. PRHR provides primary protection for non-LOCA events when the normal heat sink is lost.

PRHR outlet temperature is provided to monitor primary system heat removal during accident conditions when the steam generators are not available. PRHR provides primary protection for non-LOCA events when the normal heat sink is lost.

13, 14, 15, 16. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

BASES

LCO (continued)

An evaluation was made of the minimum number of valid core exit thermocouples necessary for In-Core Cooling (ICC) detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate ICC detection is assured with two valid core exit thermocouples per quadrant. Core Exit Temperature is also used for plant stabilization and cooldown monitoring.

Two OPERABLE channels of Core Exit Temperature are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Two thermocouples in each of the two divisions ensure a single failure will not disable the ability to determine the temperature at two locations within a quadrant.

17. <u>Passive Containment Cooling System (PCS) Storage Tank Level and PCS Flow</u>

The PCS must be capable of removing the heat from the containment following a postulated LOCA or steam line break (SLB). The tank level instruments provide indication that sufficient water is available to meet this requirement. The PCS flow instrument provides a diverse indication of the PCS heat removal capability.

18. <u>Remotely Operated Containment Isolation Valve Position</u>

The Remotely Operated Containment Isolation Valve Position is provided for verification of containment OPERABILITY.

19. <u>IRWST to RNS Suction Valve Status</u>

The position of the motor-operated valve in the line from the IRWST to the pump suction header is monitored to verify that the valve is closed following postulated events. The valve must be closed to prevent loss of IRWST inventory into the RNS.

BASES

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables provide the information necessary to assess the process of accomplishing or maintaining critical safety functions following Design Basis Accidents (DBAs). The applicable DBAs 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 that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

The ACTIONS Table has been modified by two Notes.

The first Note excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a plant shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident using alternate instruments and methods, and low probability of an event requiring these instruments.

The second Note in the ACTIONS clarifies the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of the inoperable channel(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>

When one or more Functions have one required channel which is inoperable, the required inoperable channel 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 channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels 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.

<u>B.1</u>

The Required Action directs actions to be taken in accordance with Specification 5.6.7 immediately. Each time an inoperable channel has not met Required Action A.1, and the associated Completion Time has expired, Condition B is entered.

BASES

ACTIONS (continued)

C.1

When one or more Functions have two required channels which are inoperable, (two channels inoperable in the same Function), one channel 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 instrument operation and the availability of alternate means to obtain the required information.

Continuous operation with two required channels 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 channel of the Function limits the risk that the PAM function will be in a degraded condition should an accident occur.

D.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.3-1. The applicable Condition referenced in the Table is Function dependent.

Each time an inoperable channel has not met any Required Action of Condition C, and the associated Completion Time has expired, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C are not met for the Functions in Table 3.3.3-1, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least 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.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each PAM instrumentation function in Table 3.3.3-1:

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days verifies that a gross instrumentation failure has not occurred. A CHANNEL CHECK is a

SURVEILLANCE REQUIREMENTS (continued)

comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar plant instruments located throughout the plant.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal-processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of those displays associated with the required channels of this LCO.

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop including the sensor. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. This SR is modified by a Note that excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.” RTD and Thermocouple channels are to be calibrated in place using cross-calibration techniques. The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

BASES

REFERENCES	1.	Regulatory Guide 1.97, Rev. 3, “Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” U.S. Nuclear Regulatory Commission.

B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown Workstation (RSW)

BASES

BACKGROUND

The RSW provides the control room operator with sufficient displays 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. Passive residual heat removal (PRHR), the core makeup tanks (CMTs), and the in-containment refueling water storage tank (IRWST) can be used to remove core decay heat. The use of passive safety systems allows extended operation in MODE 4.

If the control room becomes inaccessible, the operators can establish control at the RSW and place and maintain the unit in MODE 4 with T_{avg} < 350°F. The unit can be maintained safely in MODE 4 with T_{avg} < 350°F for an extended period of time.

The OPERABILITY of the remote shutdown control and display functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 4 with $T_{avg} \geq$ 350°F should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The RSW is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 4 with T_{avg} < 350°F.

The criteria governing the design and the specific system requirements of the RSW are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

Since the passive safety systems alone can establish and maintain safe shutdown conditions for the unit, nonsafety systems are not required for safe shutdown of the unit. Therefore, no credit is taken in the safety analysis for nonsafety systems.

The RSW satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The RSW LCO provides the OPERABILITY requirements of the displays and controls necessary to place and maintain the unit in MODE 4 from a location other than the control room.

The RSW is OPERABLE if the display instrument and control functions needed to support the RSW are OPERABLE.

BASES

LCO (continued)

The RSW covered by this LCO does not need to be energized to be considered OPERABLE. This LCO is intended to ensure the RSW will be OPERABLE if unit conditions require that the RSW be placed in operation.

APPLICABILITY

The RSW LCO is applicable in MODES 1, 2, and 3 and in MODE 4 with T_{avg} < 350°F. This is required so that the facility can be placed and maintained in MODE 4 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4 with T_{avg} < 350°F or in MODE 5 or 6. In these MODES, the unit is already subcritical and in a condition of reduced Reactor Coolant System (RCS) energy. Under these conditions, considerable time is available to restore necessary instrument control functions if control room instruments or controls become unavailable.

ACTIONS

The Note excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the low probability of an event requiring the RSW and because the equipment can generally be repaired during operation without significant risk of a spurious trip.

A.1

Condition A addresses the situation where the RSW is inoperable. The Required Action is to restore the RSW 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 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 at least MODE 3 within 6 hours and to MODE 4 with T_{avg} < 350°F 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 verifies that each required RSW transfer switch performs the required functions. This ensures that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 4 with T_{avg} < 350°F from the RSW. The 24 month Frequency was developed considering it is prudent that these types of surveillances be performed during a unit outage. However, this surveillance is not required to be performed only during a unit outage. This is due to the unit conditions needed to perform the surveillance and the potential for unplanned transients if the surveillance is performed with the reactor at power. Operating experience demonstrates that RSW transfer switches usually pass the surveillance test when performed on the 24 month Frequency.

SR 3.3.4.2

This Surveillance verifies that the RSW communicates controls and indications with Divisions A, B, C, and D of the PMS. Communication is accomplished by use of separate multiplexers for each division. The operator can select the controls and indications available through each PMS division.

The Frequency is based on the known reliability of the Functions and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.4.3

SR 3.3.4.3 verifies the OPERABILITY of the RSW hardware and software by performing diagnostics to show that operator displays are capable of being called up and displayed to an operator at the RSW. The RSW has several video display units which can be used by the operator. The video display units are identical to that provided in the control room and the operator can display information on the video display units in a manner which is identical to the way the information is displayed in the control room. The operator normally selects an appropriate set of displays based on the particular operational goals being controlled by the operator at the time. Each display consists of static graphical and legend information which is contained within the display processor associated with each video display unit and dynamic data which is updated by the data display system.

The Frequency of 24 months is based on the use of the data display capability in the control room as part of the normal unit operation and the

BASES

SURVEILLANCE REQUIREMENTS (continued)

availability of multiple video display units at the RSW. The Frequency of 24 months is based upon operating experience and consistency with control room hardware and software.

SR 3.3.4.4

SR 3.3.4.4 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) every 24 months. This test should verify the OPERABILITY of the reactor trip breakers (RTBs) open and closed indication on the RSW by actuating the RTBs. The Frequency of 24 months was chosen because the RTBs may not be exercised while the facility is at power and is based on operating experience and consistency with the refueling outage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.

2. Section 7.4.1, "Safe Shutdown."

B 3.3 INSTRUMENTATION

B 3.3.5 Diverse Actuation System (DAS) Manual Controls

BASES

BACKGROUND

The Diverse Actuation System (DAS) manual controls provide non-Class 1E backup controls in case of common-mode failure of the Protection and Safety Monitoring System (PMS) automatic and manual actuations evaluated in the AP1000 PRA. These DAS manual controls are not credited for mitigating accidents in the DCD Chapter 15 analyses.

The specific DAS controls were selected based on PRA risk importance as discussed in Reference 1. As noted in Reference 1, electrical power for these controls and instrument indications need not be covered by Technical Specifications. The rational is that these controls use the same nonsafety-related power supply used by the plant control system. This power is required to be available to support normal operation of the plant. With offsite power available, there are several sources to provide this power including AC power to non-Class 1E battery chargers, AC power to rectifiers, and non-Class 1E batteries. As a result, with offsite power available it is very likely that power will be available for these DAS controls. If offsite power is not available, then there is still the likelihood that the non-1E batteries or the non-1E diesel generators will be available. Even if these sources are unavailable, the desired actions will occur without operator action for the more probable events. The rods will insert automatically on loss of offsite power. The passive residual heat removal heat exchanger (PRHR HX), core makeup tanks (CMT), passive containment cooling system (PCS), and containment isolation features are initiated by operation of fail-safe, air-operated valves. If all offsite and onsite AC power is lost, the instrument air system will depressurize by the time these functions are needed in the 1-hour time frame.

Instrument readouts are expected to be available even in case of complete failure of the PMS due to common cause failure. These instruments include both DAS and PLS instruments. They are powered by DC sources for 24 or 72 hours following a loss of AC power, as described in DCD Section 8.3.2. As discussed above, it is expected that AC power will be available to power the instruments. Even if the operators have no instrument indications, they are expected to actuate the controls most likely to be needed (PRHR HX, CMT, PCS, and containment isolation). If all AC power fails, then the rods will drop and the air-operated valves will go to their fail-safe positions.

The DAS uses equipment from sensor output to the final actuated device that is diverse from the PMS to automatically initiate a reactor trip, or to manually actuate the identified safety-related equipment. DCD Section 7.7.1.11 (Ref. 2) provides a description of the DAS.

BASES

APPLICABLE SAFETY ANALYSES

The DAS manual controls are required to provide a diverse capability to manually trip the reactor and actuate the specified safety-related equipment, based on risk importance in the AP1000 PRA.

The DAS manual controls are not credited for mitigating accidents in the DCD Chapter 15 safety analyses.

The AP1000 PRA, Appendix A, provides additional information, including the thermal and hydraulic analyses of success sequences used in the PRA.

The DAS manual controls satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The DAS LCO provides the requirements for the OPERABILITY of the DAS manual trip and actuation controls necessary to place the reactor in a shutdown condition and to remove decay heat in the event that the PMS automatic actuation and manual controls are inoperable.

APPLICABILITY

The DAS manual controls are required to be OPERABLE in the MODES specified in Table 3.3.5-1.

The manual DAS reactor trip control is required to be OPERABLE in MODES 1 and 2 to mitigate the effects of an ATWS event occurring during power operation.

The other manual DAS actuation controls are required to be available in the plant MODES specified, based on the need for operator action to actuate the specified components during events that may occur in these various plant conditions, as identified in the AP1000 PRA.

ACTIONS

A.1

Condition A applies when one or more DAS manual controls are inoperable.

The Required Action A.1 to restore the inoperable DAS manual control(s) to OPERABLE status within 30 days is reasonable because the DAS is a separate and diverse non-safety backup system for the manual reactor trip and manual safety-related equipment actuation controls. The 30-day Completion Time allows sufficient time to repair an inoperable manual DAS control but ensures the control is repaired to provide backup protection.

BASES

ACTIONS (continued)

B.1 and B.2

Condition B applies when Required Action A cannot be completed for the DAS manual reactor trip control within the required completion time of 30 days.

Required Action B.1 requires SR 3.3.1.6, “Perform TADOT” for the reactor trip breakers, to be performed once per 31 days, instead of once every 92 days. Condition A of Example 1.3-6 illustrates the use of the Completion Time for Required Action B.1. The initial performance of SR 3.3.1.6 on the first division (since it is performed on a STAGGERED TEST BASIS) must be completed within 31 days of entering Condition B. The normal surveillance test frequency requirements for SR 3.3.1.6 must still be satisfied while performing SR 3.3.1.6 for Required Action B.1. The predominant failure requiring the DAS manual reactor trip control is common-mode failure of the reactor trip breakers. This change in surveillance frequency for testing the reactor trip breakers increases the likelihood that a common-mode failure of the reactor trip breakers would be detected while the DAS manual reactor trip control is inoperable. This reduces the likelihood that a diverse manual reactor trip is required. It is not required to perform a TADOT for the manual actuation control. The manual reactor trip control is very simple, highly reliable, and does not use software in the circuitry. Although the DAS manual controls are non-Class 1E, they have been shown to be PRA risk important as discussed in Reference 1. The impact of an inoperable DAS manual control is compensated for by increasing the reactor trip breaker surveillance frequency from once every 92 days to once every 31 days.

Action B.2 requires that the inoperable DAS manual reactor trip control be restored to OPERABLE status prior to entering MODE 2 following any plant shutdown to MODE 5 while the control is inoperable. This ACTION is provided to ensure that all DAS manual controls are restored to OPERABLE status following the next plant shutdown.

C.1 and C.2

Condition C applies when Required Action A cannot be completed for any DAS manual actuation control (other than reactor trip) within the required completion time of 30 days.

Required Action C.1 requires SR 3.3.2.2, “Perform ACTUATION LOGIC TEST,” to be performed once per 31 days, instead of once every 92 days. Condition A of Example 1.3-6 illustrates the use of the Completion Time for Required Action C.1. The initial performance of SR 3.3.2.2 on the first

ACTIONS (continued)

division (since it is performed on a STAGGERED TEST BASIS) must be completed within 31 days of entering Condition C. The normal surveillance test frequency requirements for SR 3.3.2.2 must still be satisfied while performing SR 3.3.2.2 for Required Action C.1. The predominant failure requiring the DAS manual actuation control is common-mode failure of the PMS actuation logic software or hardware. This change in surveillance frequency for actuation logic testing increases the likelihood that a common-mode failure of the PMS actuation logic from either cause would be detected while any DAS manual actuation control is inoperable. This reduces the likelihood that a diverse component actuation is required. It is not required to perform a TADOT for the manual actuation control device since the manual actuation control devices are very simple and highly reliable. Although the DAS manual controls are non-Class 1E, they have been shown to be PRA risk important as discussed in Reference 1. The impact of an inoperable DAS manual control is compensated for by increasing the automatic actuation surveillance frequency from once every 92 days to once every 31 days.

Action C.2 requires that the inoperable DAS manual actuation control(s) be restored to OPERABLE status prior to entering MODE 2 following any plant shutdown to MODE 5 while the control is inoperable. This ACTION is provided to ensure that all DAS manual controls are restored to OPERABLE status following the next plant shutdown.

D.1 and D.2

Condition D is entered if the Required Action associated with Condition B or C is not met within the required Completion Time.

Required Actions D.1 and D.2 ensure that the plant is placed in a condition where the probability and consequences of an event are minimized. The allowed Completion Times are reasonable based on plant operating experience, for reaching the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

SR 3.3.5.1 is the performance of a TADOT of the DAS manual trip and actuation controls for the specified safety-related equipment. This TADOT is performed every 24 months.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency is based on the known reliability of the DAS functions and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of the setpoints from the TADOT. The functions have no setpoints associated with them.

REFERENCES

1. WCAP-15985, “AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process,” Revision 2, dated August 2003.

2. DCD, Section 7.7.1.11.

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 flow rate within the 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 of operating conditions. The limits placed on RCS pressure, temperature, and 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 DNBR limits.

The RCS coolant average 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 temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. At the beginning of the first fuel cycle, precision (calorimetric) flow measurements, augmented by hydraulic measurements in the reactor coolant loop and pump performance, provide a value for comparison to the limit. The reactor coolant flow rate channels are normalized to these test measurements for 100-percent indication and are frequently monitored to determine flow degradation. A lower RCS flow 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 transients initiated within the requirements of this LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit which could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed include loss of coolant flow events and

BASES

APPLICABLE SAFETY ANALYSES (continued)

dropped or stuck rod events. An 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 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit and the RCS average temperature limit specified in the COLR correspond to analytical limits, with an allowance for steady state fluctuations and measurement errors. The RCS average temperature limit corresponds to the analytical limit with allowance for controller deadband and measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables, pressurizer pressure, RCS average 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. However, the minimum RCS flow, usually based on maximum analyzed steam generator tube plugging, is retained in the TS LCO. Operating within these limits will result in meeting DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error based on performing precision flow measurements and using the result to normalize the RCS flow rate indicators.

The numerical values for pressure, temperature, and flow rate specified in the COLR are given for the measurement location but have been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state plant operation in order to ensure DNBR criterion will be met in the event of an unplanned loss of forced coolant flow or other DNB-limiting 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 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

BASES

APPLICABILITY (continued)

counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, 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 Action A.1 is 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

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency of pressurizer pressure is sufficient to ensure the pressure can be

BASES

SURVEILLANCE REQUIREMENTS (continued)

restored to a normal operation, steady state condition following loadchanges and other expected transient operations. 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

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour Frequency 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 3.4.1.3

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 precision test measurements once every 24 months, at the beginning of each fuel cycle, allows the installed RCS flow instrumentation to be normalized and verifies the actual RCS flow is greater than or equal to the minimum required RCS flow rate. These test measurements may be based on a precision heat balance, or by differential pressure measurements of static elements in the RCS piping (such as elbows) that have been calibrated by previous precision tests, or by a combination of those two methods. In all cases, the measured flow, less allowance for error, must exceed the value used in the safety analysis and specified in the COLR.

The Frequency of 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 after 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, “Accident Analyses.”

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 zero 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 assume 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 ≥ the HZP temperature of 557°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°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 parameter satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical (k_{eff} ≥ 1.0) at a temperature less than a small band below the HZP temperature, which is 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} ≥ 1.0, LCO 3.4.2 is applicable since the reactor can only be critical (k_{eff} ≥ 1.0) in these MODES.

The special test exception of LCO 3.1.8, “MODE 2 PHYSICS TEST Exceptions,” permits PHYSICS TESTS to be performed at ≤ 5.0% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

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 551°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, "Accident Analyses."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

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 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.

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. 1) requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. An adequate margin to brittle failure must be provided during normal operation, anticipated operational occurrences, and system hydrostatic tests. Reference 1 mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

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. 3) and Appendix H of 10 CFR 50 (Ref. 4). 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. 5).

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 P/T span of the 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 1 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. ASME Code, Section XI, Appendix E (Ref. 6) 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 7 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(c)(2)(ii).

BASES

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, ISLH testing and criticality; 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 (brittle) failure in accordance with 10 CFR 50, Appendix G (Ref. 1). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, they are applicable 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

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures.

A.1 and A.2

Operation outside the P/T limits must be restored to within the limits. The RCPB must be returned to a condition that has been verified by stress analyses. Restoration is in the proper direction to reduce RCPB stress.

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 before continuing operation. Several methods may be used, including comparison with preanalyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6) 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. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 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 per Required Action A.1 alone 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 a Required Action and associated Completion Time of Condition A are 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 increased stress, or a sufficiently severe event caused entry into an unacceptable region. 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 in 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 operate pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 4 within 24 hours, with RCS pressure < 500 psig.

The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power condition in an orderly manner without challenging plant systems.

C.1 and C.2

Actions 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.

The immediate Completion Time reflects the urgency of initiating action to restore 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 that the RCPB integrity remains acceptable and must be completed prior to

BASES

ACTIONS (continued)

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. 6), 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.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 C.1 is 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 PTLR limits is required every 30 minutes when RCS P/T 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 surveillance to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2, “RCS Minimum Temperature for Criticality,” contains a more restrictive requirement.

REFERENCES

1. 10 CFR 50, Appendix G, “Fracture Toughness Requirements.”

2. ASME Boiler and Pressure Vessel Code, Section III, Appendix G, “Protection Against Non-Ductile Failure.”

3. ASTM E 185-82, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,” July 1982.

BASES

REFERENCES (continued)

4. 10 CFR 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements.”

5. “Embrittlement of Reactor Vessel Materials,” May 1988.

6. ASME Boiler and Pressure Vessel Code, Section XI, Appendix E, “Evaluation of Unanticipated Operating Events.”

7. WCAP-7924-A, “Basis for Heatup and Cooldown Limit Curves,” April 1975.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops

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. Removal of the heat generated in the fuel due to fission-product decay following a unit shutdown.

The reactor coolant is circulated through two loops connected in parallel to the reactor vessel, each containing a SG, two reactor coolant pumps (RCPs), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the primary 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.

The RCPs must be started using the variable speed controller with the reactor trip breakers open. The controller shall be bypassed prior to closure of the reactor trip breakers.

APPLICABLE SAFETY ANALYSES

MODES 1 and 2

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 and RCPs in service.

APPLICABLE SAFETY ANALYSES (continued)

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 two RCS loops are initially in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses, where RCP operation is most important 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 two RCS loop operation. For two 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% RATED THERMAL POWER (RTP). This is the design overpower condition for two RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 118% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points which result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with both 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.

<u>MODES 3, 4, and 5</u>

Whenever the reactor trip breakers are in the closed position and the control rod drive mechanisms (CRDMs) are energized, there is the possibility of an inadvertent rod withdrawal from subcritical, resulting in a power excursion in the area of the withdrawn rod. Such a transient could be caused by a malfunction of the Plant Control System (PLS). In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM. The initial power rise is terminated by doppler broadening in the fuel pins, followed by rod insertion. During this event, if there is not adequate coolant flow along the clad surface of the fuel, there is a potential to exceed the departure from nucleate boiling ratio (DNBR) limit. Therefore, the required coolant flow is an initial condition of a design basis event that presents a challenge to the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES (continued)

Therefore, in MODE 3, 4 or 5 with the RTBs in the closed position and the PLS capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires the RCPs to be OPERABLE and in operation to ensure that the accident analysis limits are met.

In MODES 3, 4 and 5 with the RTBs open, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. This is addressed in LCO 3.4.8, "Minimum RCS Flow."

RCS Loops satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(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 DNB, four pumps are required in MODES 1 and 2. The requirement that at least four RCPs must be operating in MODES 3, 4 and 5 when the RTBs are closed provides assurance that, in the event of a rod withdrawal accident, there will be adequate flow in the core to avoid exceeding the DNBR limit. Bypass of the RCP variable speed control ensures that the pumps are operating at full flow.

With the RTBs in the open position, the PLS is not capable of rod withdrawal; therefore only a minimum RCS flow of 3,000 gpm is necessary to ensure removal of decay heat from the core in accordance with LCO 3.4.8, Minimum RCS Flow.

Note 1 prohibits startup of a RCP when the reactor trip breakers are closed. This requirement prevents startup of a RCP and the resulting circulation of cold and/or unborated water from an inactive loop into the core, precluding reactivity excursion events which are unanalyzed.

Note 2 prohibits startup of an RCP when the RCS temperature is ≥ 200°F unless pressurizer level is < 92%. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 3 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ 200°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

BASES

LCO (continued)

Note 4 permits all RCPS to be de-energized in MODE 3, 4, or 5 for ≤ 1 hour per 8 hour period. The purpose of the NOTE is to permit tests that are designed to validate various accident analysis values. One of these tests is for the 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 may need to be revalidated by conducting the test again.

Another test performed during the startup testing program is the validation of the rod drop times during cold conditions, both with and without flow.

The no-flow tests may be performed in MODE 3, 4, or 5, and require that the pumps be stopped for a short period of time. The Note permits the de-energizing 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 only be performed once, unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests and 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 at boron concentrations less than required to assure the SDM of LCO 3.1.1, 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 natural circulation flow obstruction.

An OPERABLE RCS loop is composed of two OPERABLE RCPs in operation providing forced flow for heat transport and an OPERABLE SG.

BASES

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, both RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

In MODES 3, 4 and 5, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. For these purposes and because the reactor trip breakers are closed, there is the possibility of an inadvertent rod withdrawal event. Four RCPs are required to be operating in MODES 3, 4 and 5, whenever the reactor trip breakers are closed.

ACTIONS

A.1

If the requirements of the LCO are not met while in MODE 1 or 2, the Required Action is to reduce power and bring the plant to MODE 3 with the reactor trip breakers open. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

Condition A is modified by a Note which requires completion of Required Action A.1 whenever the Condition is entered. This ensures that no attempt is made to restart a pump with the reactor trip breakers closed, thus precluding events which are unanalyzed.

When all four reactor coolant pumps are operating, a loss of a single reactor coolant pump above power level P-10 will result in an automatic reactor trip.

The Completion Time of 6 hours is reasonable to allow for an orderly transition to MODE 3. The applicable safety analyses described above bound Design Basis Accidents (DBA) initiated with three reactor coolant pumps operating at power levels below P-10.

B.1

If the requirements of the LCO are not met while in MODE 3, 4 or 5, the Required Action is to remain in MODE 3, 4 or 5 and open the reactor trip breakers. This action eliminates the possibility of a rod withdrawal event with one or more pumps not operating and thus minimizing the possibility of violating DNB limits.

BASES

ACTIONS (continued)

Condition B is modified by a Note which requires completion of Required Action B.1 whenever the Condition is entered. This ensures that no attempt is made to restart a pump with the reactor trip breakers closed, thus precluding events which are unanalyzed.

The Completion Time of 1 hour is reasonable to allow for planned opening of the reactor trip breakers, since plant cool-down is not required.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours that each RCS loop is in operation with the pump variable speed control bypassed. Verification includes flow rate and temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the main control room to monitor RCS loop performance.

REFERENCES

1. Chapter 15, “Accident Analysis.”

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 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 normal level and pressure control components addressed by this LCO include the pressurizer water level, the heaters, their controls, and power supplies. Pressurizer safety valves and automatic depressurization valves are addressed by LCO 3.4.6, “Pressurizer Safety Valves,” and LCO 3.4.11, “Automatic Depressurization System (ADS) – Operating,” respectively. 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.
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 pressurizer heater operation, however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirement for the pressurizer water volume ≤ 92% of span, ensures that an adequate 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.

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.

ACTIONS

A.1 and A.2

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.

If the pressurizer water level is above the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. This is done by restoring the level to within limit, within 6 hours, or by placing the unit in MODE 3 with the reactor trip breakers open within 6 hours, and placing the unit in MODE 4 within 12 hours. This takes the unit out of the applicable MODES and restores the unit to operation within the bounds of the safety analyses.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

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 Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

REFERENCES

1. Chapter 15, “Accident Analysis.”

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 Pressurizer Safety Valves

BASES

BACKGROUND

The two pressurizer safety valves provide, in conjunction with the Protection and Safety Monitoring System (PMS), overpressure protection for the RCS. The pressurizer safety valves are totally enclosed, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2733.5 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The minimum relief capacity for each valve, 750,000 lb/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 pressurizer safety valves discharge into the containment atmosphere. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves.

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 when the reactor vessel head is on; however, in MODE 4 with the RNS aligned, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.14, “Low Temperature Overpressure Protection (LTOP) System.”

The upper and lower pressure limits are based on the ± 1% tolerance requirement (Ref. 1) 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 pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

The consequences of exceeding the ASME Code, Section III 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.

BASES

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in Chapter 15 (Ref. 3) that require safety valve actuation assume operation of two pressurizer safety valves to limit increases in the RCS pressure. The overpressure protection analysis (Ref. 2) is also based on operation of the two safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load;

d. Locked rotor; and

e. Loss of AC power/loss of normal feedwater

Detailed analyses of the above transients are contained in Reference 3. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer Safety Valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psia), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) 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 with the RNS isolated or with the RCS temperature ≥ 275°F, OPERABILITY of two 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 valves for protection.

BASES

APPLICABILITY (continued)

The LCO is not applicable in MODE 4 with RNS open and in MODE 5, because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

The Note allows entry into MODES 3 and 4 with the lift setpoints outside the LCO limits. This permits testing and examination of the safety valves 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 36 hour exception is based on 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection 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 the Required Action of A.1 cannot be met within the required Completion Time or if two pressurizer safety valves are inoperable, the plant must be placed in 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 the RNS aligned to the RCS and RCS temperature < 275°F 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 the RNS aligned to the RCS, overpressure protection is provided by the LTOP System. 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 two pressurizer safety valves.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested one at a time and in accordance with the requirements of ASME OM Code (Ref. 4), which provides the activities and Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the values are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, NB 7614.3.

2. WCAP-16779, "AP1000 Overpressure Protection Report, April 2007."

3. Chapter 15, "Accident Analyses."

4. ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core comprise the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves 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 RCS 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% leaktight. 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 a 300 gpd primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leak contaminates the secondary fluid.

The Chapter 15 (Ref. 3) analyses for the accidents involving secondary side releases assume 150 gpd primary to secondary LEAKAGE in each generator as an initial condition. The design basis radiological consequences resulting from a postulated SLB accident and SGTR are provided in Sections 15.1.5 and 15.6.3 of Chapter 15, respectively.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operation LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

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 are not pressure boundary LEAKAGE.

b. <u>Unidentified LEAKAGE</u>

0.5 gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air N13/F18 radioactivity monitoring and containment sump level monitoring equipment, can detect within a reasonable time period. This leak rate supports leak before break (LBB) criteria. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

LCO (continued)

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. Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through One 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.

e. Primary to IRWST LEAKAGE through the PRHR Heat Exchanger (HX)

The 500 gpd limit from the PRHR HX is based on the assumption that a single crack leaking this amount would not lead to a PRHR HX tube rupture under the stress condition of an RCS pressure increase event. If leaked through many cracks, the cracks are very small, and the above assumption is conservative. This is conservative because the thickness of the PRHR HX tubes is approximately 60% greater than the thickness of the SG tubes. Furthermore, a PRHR HX tube rupture would result in an isolable leak and would not lead to a direct release of radioactivity to the atmosphere.

BASES

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.

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 limits, 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 to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors which 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 ACTIONS 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 3.4.7.1

Verifying RCS LEAKAGE 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.

Unidentified LEAKAGE and identified LEAKAGE are determined by performance of a RCS water inventory balance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The RCS water inventory balance must be met with the reactor at steady state operating conditions. 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 inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, and with no makeup or letdown.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere N13/F18 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 3.4.9, “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 containment atmosphere N13/F18 radioactivity LEAKAGE measurement is valid only for plant power > 20% RTP.

The containment atmosphere N13/F18 radioactivity LEAKAGE measurement during MODE 1 is not valid while containment purge occurs or within 2 hours after the end of containment purge.

The containment sump level change method of detecting leaks during MODES 1, 2, 3, and 4 is not valid while containment purge occurs or within 2 hours after the end of containment purge.

The containment sump level change method of detecting leaks during MODES 1, 2, 3, and 4 is not valid during extremely cold outside ambient conditions when frost is forming in the interior of the containment vessel.

The 72-hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one 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 3.4.18, "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 any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

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, “Accident Analysis.”
4. NEI-97-06 – “Steam Generator Program Guidelines.”
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 Minimum RCS Flow

BASES

BACKGROUND

The AP1000 RCS consists of the reactor vessel and two heat transfer loops, each containing a steam generator (SG), two reactor coolant pumps (RCPs), a single hot leg and two cold legs for circulating reactor coolant. Loop 1 also contains connections to the pressurizer and passive residual heat removal (PRHR).

The primary function of the reactor coolant is removal of decay heat and the transfer of this heat, via the SGs to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

Within the RCS, coolant loop flow can be provided by the reactor coolant pumps, the Normal Residual Heat Removal System (RNS), and to a lesser degree when in the passive mode of operation, natural circulation.

APPLICABLE SAFETY ANALYSES

An initial condition in the Design Basis Accident (DBA) analysis of a possible Boron Dilution Event (BDE) in MODE 3, 4, or 5 is the assumption of a minimum mixing flow in the RCS. In this scenario, dilute water is inadvertently introduced into the RCS, is uniformly mixed with the primary coolant, and flows to the core. The increase in reactivity is detected by the source range instrumentation which provides a signal to terminate the inadvertent dilution before the available SDM is lost. If there is inadequate mixing in the RCS, the dilute water may stratify in the primary system, and there will be no indication by the source range instrumentation that a dilution event is in progress. When primary flow is finally increased, the dilution event may have progressed to the point that mitigation by the source range instrumentation is too late to prevent the loss of SDM.

Thus, a minimum mixing flow in the RCS is a process variable which is an initial condition in a DBA analysis.

Minimum RCS Flow satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that a minimum RCS flow be maintained provides assurance that in the event of an inadvertent BDE, the diluted water will be properly mixed with the primary system coolant, and the increase in core reactivity will be detected by the source range instrumentation.

LCO (continued)

Note 1 permits all RCPS to be de-energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analysis values. One of these tests is for the 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 may need to be revalidated by conducting the test again.

Another test performed during the startup testing program is the validation of the rod drop times during cold conditions, both with and without flow.

The no-flow tests may be performed in MODE 3, 4, or 5, and require that the pumps be stopped for a short period of time. The Note permits the de-energizing 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 only be performed once, unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests and 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 at boron concentrations less than required to assure the SDM of LCO 3.1.1, 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 natural circulation flow obstruction.

Note 2 prohibits startup of an RCP when the RCS temperature is ≥ 200°F unless pressurizer level is < 92%. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

BASES

LCO (continued)

Note 3 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ 200°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

APPLICABILITY

Minimum RCS flow is required in MODES 3, 4, and 5 with the reactor trip breakers (RTBs) open and with unborated water sources not isolated from the RCS because an inadvertent BDE is considered possible in these MODES.

In MODES 1 and 2, and in MODES 3, 4, and 5 with the RTBs closed, LCO 3.4.4 requires all four RCPs to be in operation. Thus, in the event of an inadvertent boron dilution, adequate mixing will occur.

A minimum mixing flow is not required in MODE 6 because LCO 3.9.2 requires that all valves used to isolate unborated water sources shall be secured in the closed position. In this situation, an inadvertent BDE is not considered credible.

ACTIONS

A.1

If no RCP is in operation, all sources of unborated water must be isolated within 1 hour. This action assures that no unborated water will be introduced into the RCS when proper mixing cannot be assured. The allowed Completion Time requires that prompt action be taken, and is based on the low probability of a DBA occurring during this time.

A.2

The Requirement to perform SR 3.1.1.1 (SDM verification) within 1 hour assures that if the boron concentration in the RCS has been reduced and not detected by the source range instrumentation, prompt action may be taken to restore the required SDM. The allowed Completion Time is consistent with that required of Action A.1 because the conditions and consequences are the same.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE REQUIREMENTS	<u>SR 3.4.8.1</u> This Surveillance requires verification every 12 hours that a minimum mixing flow is present in the RCS. A Frequency of 12 hours is adequate considering the low probability of an inadvertent BDE during this time, and the ease of verifying the required RCS flow.
REFERENCES	None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Leakage Detection Instrumentation

BASES

BACKGROUND	GDC 30 of Appendix A to 10CFR50 (Ref. 1) requires means for detecting, and, to the extent practical, identifying 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 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, is instrumented to alarm for increases of 0.5 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE. Note that the containment sump level instruments are also used to identify leakage from the main steam lines inside containment. Since there is not another method to identify steam line leakage in a short time frame, two sump level sensors are required to be operable. The containment water level sensors (LCO 3.3.3) provide a diverse backup method that can detect a 0.5 gpm leak within 3.5 days. The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity used for leak detection is the decay of N13/F18. The production of N13 and F18 is proportional to the reactor power level. N13 has a short half life and comes to equilibrium quickly. F18 has a longer half life and is the dominant source used for leak detection. Instrument sensitivities for gaseous monitoring are practical for these LEAKAGE detection systems. The Radiation Monitoring System includes monitoring N13/F18 gaseous activities to provide leak detection.
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 Chapter 15 (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 leak occur.

RCS LEAKAGE detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(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 small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump level monitor, in combination with an N13/F18 gaseous activity monitor, provides an acceptable minimum. Containment sump level monitoring is performed by three redundant, seismically qualified level instruments. The LCO note clarifies that if LEAKAGE is prevented from draining to the sump, its level change measurements made by OPERABLE sump level instruments will not be valid for quantifying the LEAKAGE.

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 ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are 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.

Containment sump level monitoring is a valid method for detecting LEAKAGE in MODES 1, 2, 3, and 4. The containment atmosphere N13/F18 radioactivity LEAKAGE measurement during MODE 1 is valid only for reactor power > 20% RTP. RCS inventory monitoring via the pressurizer level changes is valid in MODES 1, 2, 3, and 4 only when RCS conditions are stable, i.e., temperature is constant, pressure is constant, no makeup and no letdown.

BASES

APPLICABILITY (continued)

The containment sump level change method of detecting leaks during MODES 1, 2, 3, and 4 is not valid while containment purge occurs or within 2 hours after the end of containment purge.

The containment atmosphere N13/F18 radioactivity LEAKAGE measurement during MODE 1 is not valid while containment purge occurs or within 2 hours after the end of containment purge.

The containment sump level change method of detecting leaks during MODES 1, 2, 3, and 4 is not valid during extremely cold outside ambient conditions when frost is forming on the interior of the containment vessel.

ACTIONS

The actions are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when leakage detection channels are inoperable. This allowance is provided because in each condition other instrumentation is available to monitor for RCS LEAKAGE.

A.1 and A.2

With one of the two required containment sump level channels inoperable, the one remaining operable channel is sufficient for RCS leakage monitoring since the containment radiation provides a method to monitor RCS leakage. However, that is not the case for the steam line leakage monitoring. The remaining operable sump level monitor is adequate as long as it continues to operate properly. Continuing plant operation is expected to result in containment sump level indication increases and in periodic operation of the containment sump pump. Therefore, proper operation of the one remaining sump level sensor is verified by the operators checking the volume input to the sump (as determined by the sump level changes and discharges from the containment) to determine that it does not change significantly. A significant change is considered to be ±10 gallons per day or 33% (whichever is greater) of the volume input for the first 24 hours after this Condition is entered. The containment sump level instruments are capable of detecting a volume change of less than 2 gallons. The containment water level sensors also provide a diverse backup that can detect a 0.5 gpm leak within 3.5 days.

Restoration of two sump channels to OPERABLE status is required to regain the function in a Completion Time of 14 days after the monitor's failure. This time is acceptable, considering the frequency and adequacy of the monitoring of the change in integrated sump discharge required by Action A.1.

BASES

ACTIONS (continued)

B.1 and B.2

With two of the two required containment sump level channels inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere N13/F18 radioactivity monitor will provide indications of changes in LEAKAGE. Together with the atmosphere monitor, the periodic surveillance for RCS inventory balance, SR 3.4.7.1, 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 3.4.7.1 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). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of one sump channel to OPERABLE status is required to regain the function in a Completion Time of 72 hours after the monitor's failure. This time is acceptable, considering the frequency and adequacy of the RCS inventory balance required by Action A.1.

C.1.1, C.1.2, and C.2

With one gaseous N13/F18 containment atmosphere radioactivity-monitoring instrumentation channel inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or RCS inventory balanced, in accordance with SR 3.4.7.1, to provide alternate periodic information.

With a sample obtained and analyzed or an RCS inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the radioactivity monitor.

The 24 hours interval for grab samples or RCS inventory balance provides periodic information that is adequate to detect LEAKAGE. A Note is added allowing that SR 3.4.7.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, and makeup and letdown). 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 leak detection is available.

BASES

ACTIONS (continued)

D.1 and D.2

If a Required Action of Condition A, B or C cannot be met within the required Completion Time, the reactor 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 without challenging plant systems.

E.1

With all required monitors inoperable, no automatic means of monitoring leakage is available and plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

SR 3.4.9.1 requires the performance of a CHANNEL CHECK of the containment atmosphere N13/F18 radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and risk and is reasonable for detecting off normal conditions.

SR 3.4.9.2

SR 3.4.9.2 requires the performance of a CHANNEL OPERATIONAL TEST (COT) on the atmosphere N13/F18 radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers risks and instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.9.3 and SR 3.4.9.4

These SRs require the performance of a CHANNEL 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.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45, “Reactor Coolant Pressure Boundary LEAKAGE Detection Systems,” U.S. Nuclear Regulatory Commission.
3. Chapter 15, “Accident Analysis.”

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 RCS Specific Activity

BASES

BACKGROUND

The limits on RCS specific activity ensure that the doses due to postulated accidents are within the doses reported in Chapter 15.

The RCS specific activity LCO limits the allowable concentration of iodines and noble gases in the reactor coolant. The LCO limits are established to be consistent with a fuel defect level of 0.25 percent and to ensure that plant operation remains within the conditions assumed for shielding and Design Basis Accident (DBA) release analyses.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to limit the doses due to postulated accidents to within the values calculated in the radiological consequences analyses (as reported in Chapter 15).

APPLICABLE SAFETY ANALYSES

The LCO limits on the reactor coolant specific activity are a factor in accident analyses that assume a release of primary coolant to the environment either directly as in a Steam Generator Tube Rupture (SGTR) or indirectly by way of LEAKAGE to the secondary coolant system and then to the environment (the Steam Line Break).

The events which incorporate the LCO values for primary coolant specific activity in the radiological consequence analysis include the following:

Steam generator tube rupture (SGTR)
Steam line break (SLB)
Locked RCP rotor
Rod ejection
Small line break outside containment
Loss of coolant accident (LOCA) (early stages)

The limiting event for release of primary coolant activity is the SLB. The SLB dose analysis considers the possibility of a pre-existing iodine spike (in which case the maximum LCO of 60 μCi/gm DOSE EQUIVALENT I-131 is assumed) as well as the more likely initiation of an iodine spike due to the reactor trip and depressurization. In the latter case, the LCO of 1.0 μCi/gm DOSE EQUIVALENT I-131 is assumed at the initiation of the accident, but the primary coolant specific activity is assumed to increase with time due to the elevated iodine appearance rate in the coolant. The reactor coolant noble gas specific activity for both cases is

BASES

APPLICABLE SAFETY ANALYSES (continued)

assumed to be the LCO of 280 μCi/gm DOSE EQUIVALENT XE-133. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.4, "Secondary Specific Activity."

The LCO limits ensure that, in either case, the doses reported in Chapter 15 remain bounding.

The RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the specific noble gas activity is limited to 280 μCi/gm DOSE EQUIVALENT XE-133. These limits ensure that the doses resulting from a DBA will be within the values reported in Chapter 15. Secondary coolant activities are addressed by LCO 3.7.4, "Secondary Specific Activity."

The SLB and SGTR accident analyses (Refs. 1 and 2) show that the offsite doses are within acceptance limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SLB or SGTR accident, lead to doses that exceed those reported Chapter 15.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity are necessary to contain the potential consequences of a SGTR to within the calculated site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to verify that DOSE EQUIVALENT I-131 is ≤ 60 μCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is to continue to provide a trend.

ACTIONS (continued)

The DOSE EQUIVALENT I-131 must be restored to normal within 48 hours. If the concentration cannot be restored to within the LCO limit in 48 hours, it is assumed that the LCO violation is not the result of normal iodine spiking.

A Note to the Required Action of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception 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.

B.1 and B.2

With DOSE EQUIVALENT XE-133 in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The allowed Completion Time of 4 hours is required to obtain and analyze a sample.

The change to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the set points of the main steam safety valves, and prevents venting the SG to the environment in a SGTR event. 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, without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is > 60 μCi/gm., the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operation experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SR 3.4.10.1 requires performing a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This is a quantitative measure of radionuclides with half lives longer than

BASES

SURVEILLANCE REQUIREMENTS (continued)

15 minutes. This Surveillance provides an indication of any increase in the release of noble gas activity from fuel rods containing cladding defects.

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 significant increase in fuel defect level during the time.

SR 3.4.10.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when increased releases of iodine from the fuel (iodine spiking) is apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level. The Frequency, between 2 and 6 hours after a power change of ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failures; samples at other times would provide inaccurate results.

REFERENCES

1. Section 15.1.5, “Steam System Piping Failure.”
2. Section 15.6.3, “Steam Generator Tube Rupture.”

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Automatic Depressurization System (ADS) – Operating

BASES

BACKGROUND

The ADS is designed to assure that core cooling and injection can be achieved for Design Basis Accidents (DBA). The four stages of ADS valves are sequenced in coordination with the passive core cooling system injection performance characteristics.

The ADS consists of 10 flow paths arranged in four different stages that open sequentially (Ref. 1). Stages 1, 2, and 3 each include 2 flow paths. Each of the stage 1, 2, 3 flow paths has a common inlet header connected to the top of the pressurizer. The outlets of the stage 1, 2, 3 flow paths combine into one of the two common discharge lines to the spargers located in the incontainment refueling water storage tank (IRWST). The first stage valves are 4 inch valves with DC motor operators. The second and third stage valves are 8 inch valves with DC motor operators. An OPERABLE stage 1, 2, or 3 automatic depressurization flow path consists of two OPERABLE normally closed motor operated valves, in series.

Stage 4 includes 4 flow paths. The fourth stage ADS valves are squib valves. The four fourth stage flow paths connect directly to the top of the reactor coolant hot legs and vent directly into the associated steam generator compartment. An OPERABLE stage 4 flow path consists of a open motor operated valve and an OPERABLE closed squib valve. These motor operated valves are not required to be OPERABLE because they are open.

The automatic depressurization valves are designed to open automatically when actuated, and to remain open for the duration of any automatic depressurization event. The valves are actuated sequentially. The stage 1 valves are actuated on a low core makeup tank (CMT) level. Stages 2 and 3 are actuated on the stage 1 signal plus time delays. Stage 4 is actuated on a Low 2 CMT level signal with a minimum time delay after stage 3. Stage 4 is blocked from actuating at normal RCS pressure.

In order to perform a controlled, manual depressurization of the RCS, the valves are opened starting with the first stage. The first stage valves can also be modulated to perform a partial RCS depressurization if required. ADS stage 1, 2, 3 valves may be manually operated under controlled conditions for testing purposes.

BASES

BACKGROUND (continued)

ADS stages 1, 2 and 3 valves are designed to open relatively slowly, from approximately 40 seconds for the first stage valves, to approximately 100 seconds for the second and third stage valves.

The ADS valves are powered by batteries. In the unlikely event that offsite and onsite AC power is lost for an extended period of time, a timer will actuate ADS within 24 hours of the time at which AC power is lost, before battery power has been degraded to the point where the valves cannot be opened.

The number and capacity of the ADS flow paths are selected so that adequate safety injection is provided from the accumulators, IRWST and containment recirculation for the limiting DBA loss of coolant accident (LOCA). For small break LOCAs the limiting single failure is the loss of one fourth stage flow path (Ref. 2). The PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of up to seven (all ADS stage 1 to 3 and one ADS stage 4) flow paths. The ADS PRA success criteria following a LOCA or non-LOCA with failure of other decay heat removal features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1, 2, 3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation.

APPLICABLE SAFETY ANALYSES

For non-LOCA events, use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the CMTs may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation.

For events which involve a loss of primary coolant inventory, such as a LOCA, the ADS will be actuated, allowing for injection from the accumulators, the IRWST, and the containment recirculation (Ref. 2).

The ADS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that the 16 ADS valves be OPERABLE ensures that upon actuation, the depressurization of the RCS will proceed smoothly and completely, as assumed in the DBA safety analyses.

For the ADS to be considered OPERABLE, the 16 ADS valves must be closed and OPERABLE (capable of opening on an actuation signal). In addition, the stage 4 motor operated isolation valves must be open. These stage 4 motor operated isolation valves are not required to be OPERABLE because they are maintained open per SR 3.4.11.1.

BASES

APPLICABILITY

In MODES 1, 2, 3 and 4 the ADS must be OPERABLE to mitigate the potential consequences of any event which causes a reduction in the RCS inventory, such as a LOCA.

The requirements for the ADS in MODES 5 and 6 are specified in LCO 3.4.12, “Automatic Depressurization System (ADS) – Shutdown, RCS Intact,” and LCO 3.4.13, “Automatic Depressurization System – Shutdown, RCS Open.”

ACTIONS

A.1

If any one flow path is determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function as long as a single failure does not also occur. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is reasonable based on the capability of the remaining ADS valves to perform the required safety functions assumed in the safety analyses and the low probability of a DBA during this time period. This Completion Time is the same as is used for two train ECCS systems which are capable of performing their safety function without a single failure.

B.1

If two flow paths, consisting of one stage 1 and either one stage 2 or 3, are determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function as long as a single failure does not also occur. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is reasonable based on the capability of the remaining ADS valves to perform the required safety functions assumed in the safety analyses and the low probability of a DBA during this time period. This Completion Time is the same as is used for two train ECCS systems which are capable of performing their safety function without a single failure.

C.1 and C.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.11 are not met for reasons other than Condition A or B, the plant must be brought to MODE 5 where the probability and consequences on an event are minimized. 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, without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Each stage 4 ADS isolation motor operated valve must be verified to be open every 12 hours. Note that these valves receive confirmatory open signals. The Surveillance Frequency is acceptable considering valve position is manually monitored in the control room.

SR 3.4.11.2

This Surveillance requires verification that each ADS stage 1, 2, 3 valve strokes to its fully open position. Note that this surveillance is performed during shutdown conditions.

The Surveillance Frequency for demonstrating valve OPERABILITY references the Inservice Testing Program.

SR 3.4.11.3

This Surveillance requires verification that each ADS stage 4 squib valve is OPERABLE in accordance with the Inservice Testing Program. The Surveillance Frequency for verifying valve OPERABILITY references the Inservice Testing Program.

The squib valves will be tested in accordance with the ASME OM Code. The applicable ASME OM Code squib valve requirements are specified in paragraph 4.6, Inservice Tests for Category D Explosively Actuated Valves. The requirements include actuation of a sample of the installed valves each 2 years and periodic replacement of charges.

REFERENCES

1. Section 6.3, “Passive Core Cooling System.”
2. Section 15.6, “Decrease in Reactor Coolant Inventory.”
3. AP1000 Probabilistic Risk Assessment, Appendix A.
4. Section 3.9.6, “Inservice Testing of Pumps and Valves.”

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Automatic Depressurization System (ADS) – Shutdown, RCS Intact

BASES

BACKGROUND	A description of the ADS is provided in the Bases for LCO 3.4.11, "Automatic Depressurization System (ADS) – Operating."
APPLICABLE SAFETY ANALYSES	For postulated events in MODE 5 with the RCS pressure boundary intact, the primary protection is the Passive Residual Heat Removal Heat Exchanger (PRHR HX). Use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the core makeup tanks (CMTs) may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation. No LOCAs are postulated during plant operation in MODE 5, however loss of primary coolant through LEAKAGE or inadvertent draining may occur. For such shutdown events occurring in MODE 5 it is anticipated that the ADS will be actuated, allowing injection from the in-containment refueling water storage tank (IRWST) and the containment recirculation if containment flooding occurs (Ref. 2). The ADS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The requirement that 9 ADS flow paths be OPERABLE assures that upon actuation, the depressurization of the RCS will proceed smoothly and completely, as assumed in the DBA safety analyses. An ADS stage 1, 2, or 3 flow path is considered OPERABLE if both valves in the line are closed and OPERABLE (capable of opening on an actuation signal). In addition, an ADS stage 4 flow path is OPERABLE if the motor operated isolation valve is open and the squib valve is closed and OPERABLE (capable of opening on an actuation signal).
APPLICABILITY	In MODE 5 with the reactor coolant pressure boundary (RCPB) intact, 9 flow paths of the ADS must be OPERABLE to mitigate the potential consequences of any event which causes a reduction in the RCS inventory, such as a LOCA.

BASES

APPLICABILITY (continued)

The requirements for the ADS in MODES 1 through 4 are specified in LCO 3.4.11, "Automatic Depressurization System (ADS) – Operating;" and in MODE 5 with the RCS pressure boundary open and MODE 6 in LCO 3.4.13, "Automatic Depressurization System (ADS) – Shutdown, RCS Open."

ACTIONS

A.1

If any one flow path is determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is acceptable since the OPERABLE ADS paths can mitigate shutdown events without a single failure.

B.1

If two flow paths, consisting of one stage 1 and either one stage 2 or 3, are determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function. A flow path is inoperable if one or two of the ADS valves in the flow path are determined to be inoperable. A Completion Time of 72 hours is acceptable since the OPERABLE ADS paths can mitigate shutdown events without a single failure.

C.1

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.12 are not met for reasons other than Condition A, the plant must be placed in a MODE in which this LCO does not apply. Action must be initiated, immediately, to place the plant in MODE 5 with the RCS pressure boundary open and ≥ 20% pressurizer level.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

The LCO 3.4.11 Surveillance Requirements are applicable to the ADS valves required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.4.11 for a discussion of each SR.

BASES

REFERENCES

1. AP1000 Probabilistic Risk Assessment, Appendix A.
2. Section 19E.4, "Safety Analyses and Evaluations."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 Automatic Depressurization System (ADS) – Shutdown, RCS Open

BASES

BACKGROUND	A description of the ADS is provided in the Bases for LCO 3.4.11, "Automatic Depressurization System (ADS) – Operating."
APPLICABLE SAFETY ANALYSES	When the plant is shutdown with the RCS depressurized, the core makeup tanks (CMTs) are isolated to prevent CMT injection. Since the ADS is actuated by low CMT level, automatic actuation of the ADS is not available. The required ADS stage 1, 2, and 3 vent paths are opened and two ADS stage 4 flow paths are OPERABLE to ensure that in-containment refueling water storage tank (IRWST) injection and containment recirculation can occur, if needed to mitigate events requiring RCS makeup, boration or core cooling (Ref. 1). The ADS vent path must be maintained until the upper internals are removed, providing an adequate vent path for IRWST injection. The ADS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The requirement that ADS stage 1, 2, and 3 flow paths be open, from the pressurizer through the spargers into the IRWST, and that two ADS stage 4 flow paths be OPERABLE assures that sufficient vent area is available to support IRWST injection. The Note allows closure of the RCS pressure boundary when the pressurizer level is < 20% to facilitate vacuum refill following mid-loop operations to establish a pressurizer water level ≥ 20%. Prior to closure of the ADS valves, compliance with LCO 3.4.12, ADS – Shutdown, RCS Intact, should be verified.
APPLICABILITY	In MODE 5 with the reactor coolant system pressure boundary (RCPB) open or pressurizer level < 20% and in MODE 6 with the upper internals in place, the stage 1, 2, and 3 ADS vent paths must be open and two ADS stage 4 flow paths be OPERABLE. The requirements for the ADS in MODES 1 through 4 are specified in LCO 3.4.11, "Automatic Depressurization System (ADS) – Operating;" and in MODE 5 with the RCPB intact in LCO 3.4.12, "Automatic Depressurization System (ADS) – Shutdown, RCS Intact."

ACTIONS

A.1 and A.2

If one required ADS stage 1, 2, or 3 flow path is closed, action must be taken to open the affected path or establish an alternative flow path within 72 hours. In this Condition the remaining open ADS stage 1, 2, and 3 flow paths and the OPERABLE ADS stage 4 flow paths are adequate to perform the required safety function without an additional single failure. The stage 4 valves would have to be opened by the operator in case of an event in this MODE. The required vent area may be restored by opening the affected ADS flow path or an alternate vent path with an equivalent area. Considering that the required function is available in this Condition a Completion Time of 72 hours is acceptable.

B.1 and B.2

If one required ADS stage 4 flow path is closed and inoperable, action must be taken to establish an alternative flow path, or restore at least two stage 4 flow paths to OPERABLE status within 36 hours. In this Condition the remaining open ADS stage 1, 2, and 3 flow paths and the one OPERABLE ADS stage 4 flow path are adequate to perform the required safety function without an additional single failure. The required vent area may be restored by opening an alternate vent path with an equivalent area. Alternatively, two stage 4 flow paths may be restored to OPERABLE status. Therefore a Completion Time of 36 hours is considered acceptable.

C.1 and C.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.13 are not met for reasons other than Conditions A or B while in MODE 5, the plant must be placed in a condition which minimizes the potential for requiring ADS venting and IRWST injection. The time to RCS boiling is maximized by increasing RCS inventory to ≥ 20% pressurizer level and maintaining RCS temperature as low as practical.

Additionally, action to suspend positive reactivity additions is required to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

D.1 and D.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.13 are not met for reasons other than Conditions A or B while in MODE 6, the plant must be placed in a

BASES

ACTIONS (continued)

condition which precludes the need for the ADS vent paths. Action must be initiated, immediately, to remove the upper internals, providing the required vent path. The time to RCS boiling is maximized by increasing RCS inventory and maintaining RCS temperature as low as practical. Additionally, action to suspend positive reactivity additions is required to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Each required ADS flow path is verified to be open by verifying that the stage 1, 2, and 3 valves are in their fully open position every 12 hours, as indicated in the control room. This Surveillance Frequency is acceptable based on administrative controls which preclude repositioning the valves.

SR 3.4.13.2

The LCO 3.4.11 Surveillance Requirements (SR 3.4.11.1 and SR 3.4.11.3) are applicable to the stage 4 ADS valves required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.4.11 for a discussion of each SR.

REFERENCES

1. Section 19E.4, “Safety Analyses and Evaluations.”

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System limits RCS pressure at low temperatures so that 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 limits which set the maximum allowable setpoints for the Normal Residual Heat Removal System (RNS) suction relief valve. LCO 3.4.3 provides 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 maximum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires isolating the accumulators. The pressure relief capacity requires the RNS suction relief valve or a depressurized RCS and an RCS vent of sufficient size. The RNS suction relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

<u>RNS Suction Relief Valve Requirements</u>

During the LTOP MODES, the RNS system is operated for decay heat removal. Therefore, the RNS suction isolation valves are open in the

BASES

BACKGROUND (continued)

piping from the RCS hot legs to the inlet of the RNS system. While these valves are open, the RNS suction relief valve is exposed to the RCS and able to relieve pressure transients in the RCS.

The RNS suction relief valve is a spring loaded, water relief valve with a pressure tolerance and an accumulation limit established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.

The RNS suction isolation valves must be open to make the RNS suction relief valves OPERABLE for RCS overpressure mitigation.

RCS Vent Requirements

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, it may require removing one or more pressurizer safety valves or manually opening one or more Automatic Depressurization System (ADS) valves. 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 the RCS temperature above 275°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. When the RNS is aligned and open to the RCS, overpressure protection is provided by the RNS suction relief valve, or a depressurized RCS and a sufficiently sized open RCS vent.

The actual temperature at which the pressure in the P/T limit curve falls below the suction relief 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 the RNS suction relief valve, or the depressurized and vented RCS condition.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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. The events listed below were used in the analysis to size the RNS suction relief valve. Therefore, any events with a mass or heat input greater than the listed events cannot be accommodated and must be prevented.

<u>Mass Input</u>

a. Makeup water flow rate to the RCS assuming both CVS makeup pumps are in operation and letdown is isolated.

<u>Heat Input</u>

a. Restart of one reactor coolant pump (RCP) with water in the steam generator secondary side 50°F hotter than the primary side water, and the RCS water solid.

<u>RNS Suction Relief Valve Performance</u>

Since the RNS suction relief valve does not have a variable P/T lift setpoint, the analysis must show that with chosen setpoint, the relief valve will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the minimum of either the P/T limit curve or 110 percent of the design pressure of the normal residual heat removal system. The current analysis shows that up to a temperature of 70°F, the mass input transient is limiting, and above this temperature the heat input transient is limiting.

To prevent the possibility of a heat input transient, and thereby limit the required flow rate of the RNS suction relief valve, an administrative requirement has been imposed that does not allow an RCP to be started with the pressurizer water level above 92% and the RCS temperature above 200°F. Under these imposed conditions, the transient created by the startup of an RCP when the RCS temperature is above 200°F can be accommodated without additional pressure relief.

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS Vent Performance

With the RCS depressurized, a vent size of 4.15 square inches is capable of mitigating a limiting overpressure transient. The area of the vent is equivalent to the area of the inlet pipe to the RNS suction relief valve so the capacity of the vent is greater than the flow possible with either the mass or heat input transient, while maintaining the RCS pressure less than the minimum of either the maximum pressure on the P/T limit curve or 110 percent of the design pressure of the normal residual heat removal system.

The required vent area may be obtained by opening one ADS Stage 2, 3, or 4 flow path.

The RCS vent size will be reevaluated 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(c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the maximum coolant input and minimum 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 input capability, the LCO requires all accumulator discharge isolation valves closed and immobilized, when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed in the PTLR.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. One OPERABLE RNS suction relief valve; or

   An RNS suction relief valve is OPERABLE for LTOP when both RNS suction isolation valves in one flow path are open, its setpoint is within limits, and testing has proven its ability to open at this setpoint.

BASES

LCO (continued)

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of ≥ 4.15 square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any cold leg temperature is below 275°F, MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 275°F. In MODE 6, the reactor vessel head is off, and overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.6, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 with the RNS isolated or RCS temperature ≥ 275°F.

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 with little or no time for operator action to mitigate the event.

The Applicability is modified by a Note stating 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.

ACTIONS

A.1, B.1, and B.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 B.1 and Required Action B.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS

ACTIONS (continued)

temperature to > 275°F, the accumulator pressure 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.

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.

C.1 and C.2

If the RNS suction relief valve is inoperable and the RCS is not depressurized, there is a potential to overpressurize the RCS and exceed the limits allowed in LCO 3.4.3. The suction relief valve is considered inoperable if the RNS isolation valves have isolated the RNS from the RCS in such a way that the suction relief valve cannot perform its intended safety function, or if the valve itself will not operate to perform its intended safety function.

Under these conditions, Required Actions C.1 or C.2 provide two options, either of which must be accomplished in 12 hours. If the RNS suction relief valve cannot be restored to OPERABLE status, the RCS must be depressurized and vented with a RCS vent which provides a flow area sufficient to mitigate any of the design low temperature overpressure events.

The 12 hour Completion Time represents a reasonable time to repair the relief valve, open the RNS isolation valves or otherwise restore the system to OPERABLE status, or depressurize and vent the RCS, without imposing a lengthy period when the LTOP system is not able to mitigate a low temperature overpressure event.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

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 main control room to verify the required status of the equipment.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.2

The RNS suction relief valve shall be demonstrated OPERABLE by verifying two RNS suction isolation valves in one flow path are open. This Surveillance is only performed if the RNS suction relief valve is being used to satisfy this LCO.

The RNS suction isolation valves are verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RNS suction isolation valves remain open.

SR 3.4.14.3

The RCS vent of ≥ 4.15 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 or a removed pressurizer safety valve or open manway also fits this category).

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.14b.

SR 3.4.14.4

The RNS suction relief valve shall be demonstrated OPERABLE by verifying that two RNS suction isolation valves in one flow path are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.14.2 for the RNS suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RNS suction relief valve is being used to meet this LCO. The ASME OM Code (Ref. 5) test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

BASES

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operation."

3. ASME Boiler and Pressure Vessel Code, Section III.

4. Section 5.2.2, "Overpressure Protection."

5. ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Pressure Isolation Valve (PIV) Integrity

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 the RCS pressure boundary as all those pressure containing components such as pressure vessels, piping, pumps, and valves which are connected to the reactor coolant system, up to and including the outermost containment isolation valve in system piping which penetrates primary reactor containment, the second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment, and the reactor coolant system safety and relief valves. This includes 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 experience varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The AP1000 PIVs are listed in Chapter 3, Table 3.9-18. The RCS PIV Leakage LCO allows RCS high pressure operation when PIV leakage has been verified.

The purpose of this specification is to prevent overpressure failure or degradation of low pressure portions of connecting systems. The following criteria was used in identifying PIVs for inclusion in the specification. A valve was included in this specification if its failure may result in:

1. Failure of low pressure portions of connected systems, such as a Loss of Coolant Accident (LOCA) outside of containment, which could place the plant in an unanalyzed condition.

2. Degradation of low pressure portions of connected systems, such as damage to a core cooling system, which could degrade a safety related function that mitigates a DBA.

Valves considered for inclusion in this specification are used to isolate the RCS from the following connected systems:

a. Passive Core Cooling System (PXS) Accumulators;

b. Normal Residual Heat Removal System (RNS); and

c. Chemical and Volume Control System (CVS).

BASES

BACKGROUND (continued)

The RNS pressure boundary isolation valves are considered to meet the first criterion for inclusion in this specification. The PXS accumulator check valves were determined to meet the second PIV criteria for inclusion in this specification. It is determined that the CVS PIVs do not meet either criteria for inclusion in this specification.

The PIVs that are addressed by this specification are listed in Chapter 3, Table 3.9-18.

The CVS pressure isolation valves were not included in this specification based on the defined criteria. The justification for excluding the CVS PIVs is discussed in the following paragraph.

The CVS contains four high pressure/low pressure connections with the RCS. Since the portion of the CVS which is located inside reactor containment is designed to full RCS pressure, the high pressure/low pressure interfaces with the RCS are the lines that penetrate the reactor containment. The CVS lines that penetrate containment include the makeup line, the letdown line to the Liquid Radwaste System, the hydrogen supply line, and the demineralizer resin sluice line used to transfer spent resins from the demineralizers to the Solid Radwaste System. These lines each contain two safety related containment isolation valves which are addressed by the Containment Isolation Specification (LCO 3.6.3). In addition to the containment isolation valves in each of the CVS lines that interface with the RCS, there are additional valves in each line that provide diverse isolation capability. Since more restrictive requirements are imposed by LCO 3.6.3, the CVS isolation valves are not included in this LCO.

Since the purpose of this LCO is to verify that the PIVs have not suffered gross failures, the valve leakage test in conjunction with tests specified in the IST program provide an acceptable method of determining valve integrity. The ability of the valves to transition from open to closed provides assurance that the valve can perform its pressure isolation function as required. A small amount leakage through these valves is allowed, provided that the integrity of the valve was demonstrated.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system or the failure of a safety related function to mitigate a DBA.

BASES

APPLICABLE SAFETY ANALYSES

Pressure isolation valve integrity is not considered in any design basis accident analyses. This specification provides for monitoring the condition of the reactor coolant pressure boundary to detect degradation which could lead to accidents or which could impair a connected system's ability to mitigate DBAs.

RCS PIV integrity satisfies, Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually small. 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 inch nominal valve size up to a maximum of 5 gpm per valve. This limit is well within the makeup capability of the CVS makeup pumps. This leak rate will not result in the overpressure of a connected low pressure system. Reference 5 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 of the valve. In such cases, the observed leakage rate at lower differential pressures can be assumed to be the leakage at the maximum pressure differential. Verification that the valve leakage diminishes with increasing pressure differential is sufficient to verify that the valve characteristics are such that higher service pressure results in a decrease in overall leakage.

APPLICABILITY

In MODES 1, 2, and 3 and MODE 4, with RCS not being cooled by the RNS, this LCO applies when the RCS is pressurized.

In MODE 4, with RNS in operation, and MODES 5 and 6, the RCS pressure is reduced and is not sufficient to overpressurize the connected low pressure systems.

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 pressurization may have affected system OPERABILITY, or isolation of an affected flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

BASES

ACTIONS (continued)

A.1

With one or more PIVs inoperable, the affected flow path(s) must be isolated. Required Action A.1 is modified by a Note that the valves used for isolation must meet the same integrity 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 8 hours. Eight hours provides time to verify IST compliance for the alternate isolation valve and isolate the flow path. The 8 hour Completion Time allows the actions and restricts the operation with inoperable isolation valves.

A.2

Required Action A.2 specifies verification that a second OPERABLE PIV can meet the leakage limits. This valve is required to be a check valve, or a closed valve, if it isolates a line that penetrates containment. For the accumulator valves, the normally open accumulator isolation valve is a suitable replacement PIV, but can remain open because leakage into the accumulator is continuously monitored. If leakage into the accumulators increased to the allowable operational leakage limit, then the valve could be used to isolate the accumulators from the RCS.

The 72 hour Completion Time allows the actions and restricts the operation with inoperable isolation valves.

B.1 and B.2

If PIV integrity cannot be restored, the system isolated, or the other Required Actions accomplished, 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 reduces the potential for a LOCA outside containment.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking

BASES

SURVEILLANCE REQUIREMENTS (continued)

valve. The leakage limit of 0.5 gpm per inch nominal valve size up to a minimum of 5 gpm 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 shall be performed every 24 months, a typical refueling cycle. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 4) as contained in the Inservice Testing Program and is within frequency allowed by the American Society of Mechanical Engineers (ASME) OM Code (Ref. 5).

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. 10 CFR 50.55a(g).
5. ASME OM Code, “Code for Operation and Maintenance of Nuclear Power Plants.”

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 Reactor Vessel Head Vent (RVHV)

BASES

BACKGROUND

The reactor vessel head vent (RVHV) is designed to assure that long-term operation of the Core Makeup Tanks (CMTs) does not result in overfilling of the pressurizer during Condition II Design Basis Accidents (DBAs). The RVHV can be manually actuated by the operators in the main control room to reduce the pressurizer water level during long-term operation of the CMTs.

The RVHV consists of two parallel flow paths each containing two RVHV isolation valves in series. The RVHV valves are connected to the reactor vessel head via a common line. The outlets of the RVHV flow paths combine into one common discharge line which connects to a single ADS discharge header that discharges to spargers located in the incontainment refueling water storage tank (IRWST). The RVHV valves are 1 inch valves with DC solenoid operators.

The RVHV valves are designed to open when actuated by the operator, and to reclose when actuated by the operator from the main control room.

The number and capacity of the RVHV flow paths are selected so that letdown flow from the RCS is sufficient to prevent pressurizer overfill for events where extended operation of the CMTs causes the pressurizer water level to increase. Although realistic evaluations of the Condition II non-LOCA events does not result in pressurizer overfill, conservative analyses of some of these events can result in pressurizer overfill if no operator actions are assumed.

APPLICABLE SAFETY ANALYSES

For Condition II non-LOCA events, such as inadvertent passive core cooling system operation and chemical and volume control system malfunction, the use of the RVHV may be required to prevent long-term pressurizer overfill (Ref. 1).

For LOCA events, the RVHV is not required.

The RVHV satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that all four RVHV valves be OPERABLE ensures that upon actuation, the RVHV can reduce the pressurizer water level as assumed in the DBA safety analyses.

BASES

LCO (continued)

For the RVHV to be considered OPERABLE, all four valves must be closed and OPERABLE (capable of opening from the main control room).

APPLICABILITY

In MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS, the RVHV must be OPERABLE to mitigate the potential consequences of any event which causes an increase in the pressurizer water level that could otherwise result in overfilling of the pressurizer.

In MODE 4, with the RCS being cooled by the RNS, and in MODES 5 and 6, operation of the CMTs or CVS will not result in a pressurizer overfill event.

ACTIONS

A.1

If one or two RVHV valves in a single flow path are determined to be inoperable, the flow path is inoperable. The remaining OPERABLE RVHV flow path is adequate to perform the required safety function. A Completion Time of 72 hours is acceptable since the OPERABLE RVHV paths can mitigate DBAs without a single failure.

B.1

If both flow paths are determined to be inoperable, the RVHV is degraded such that the system is not available for some DBA non-LOCA analyses for which it may be required. A Completion Time of 6 hours is permitted to restore at least one flow path. This Completion Time is acceptable considering that the realistic analysis of these non-LOCA events do not result in pressurizer overfill.

C.1 and C.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.16 are not met for reasons other than Conditions A or B, the plant must be brought to MODE 4 where the probability and consequences of an event are minimized. 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, without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

The dedicated component level remote manual valve switches in the main control room shall be used to stroke each RVHV valve to demonstrate OPERABILITY of the controls.

This Surveillance requires verification that each RVHV valve strokes to its fully open position. The Surveillance Frequency for demonstrating valve OPERABILITY references the Inservice Testing Program.

REFERENCES

1. Section 15.5, "Increase in Reactor Coolant System Inventory."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Chemical and Volume Control System (CVS) Makeup Isolation Valves

BASES

BACKGROUND	One of the principle functions of the CVS system is to maintain the reactor coolant inventory by providing water makeup for reactor coolant system (RCS) LEAKAGE, shrinkage of the reactor coolant during cooldowns, and RCS boron concentration changes. In the automatic makeup mode of operation, the pressurizer water level starts and stops CVS makeup to the RCS. Although the CVS is not considered a safety related system, certain isolation functions of the system are considered safety related functions. The appropriate isolation valves have been classified and designed as safety related. One of the safety related functions provided by the CVS is the termination of RCS makeup to prevent overfilling of the pressurizer during non-LOCA transients or to prevent steam generator overfilling during a steam generator tube rupture. The CVS makeup line containment isolation valves provide this RCS makeup isolation function.
APPLICABLE SAFETY ANALYSES	One of the initial assumptions in the analysis of several non-LOCA events and during a steam generator tube rupture accident is that excessive CVS makeup to the RCS may aggravate the consequences of the accident. The need to isolate the CVS makeup to the RCS is detected by the pressurizer level instruments or the steam generator narrow range level instruments. These instruments will supply a signal to the makeup line containment isolation valves in the CVS causing these valves to close and terminate RCS makeup. Thus the CVS makeup isolation valves are components which function to mitigate an accident. CVS isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The requirement that at least two CVS makeup isolation valves be OPERABLE assures that there will be redundant means available to terminate CVS makeup to the RCS during a non-LOCA event or a steam generator tube rupture accident should that become necessary.
APPLICABILITY	The requirement that at least two CVS makeup isolation valves be OPERABLE is applicable in MODES 1, 2, 3, and 4 with the normal residual heat removal system (RNS) suction to the RCS not open

BASES

APPLICABILITY (continued)

because a pressurizer overfill event or steam generator tube rupture accident is considered possible in these MODES, and the automatic closure of these valves is assumed in the safety analysis.

In the applicable MODES, the need to isolate the CVS makeup to the RCS is detected by the pressurizer level instruments (high 1 setpoint coincident with safeguards actuation or high 2 setpoint) or the steam generator narrow range level instruments (high 2 setpoint).

This isolation function is not required in MODE 4 with the RNS suction open to the RCS or in lower MODES. In such MODES, pressurizer or steam generator overfill is prevented by the RNS suction relief valve.

ACTIONS

A.1

If only one CVS makeup isolation valve is OPERABLE, the second valve must be restored to OPERABLE status in 72 hours. The allowed Completion Time assures expeditious action will be taken, and is acceptable because the safety function of automatically isolating RCS makeup can be accomplished by the redundant isolation valve.

B.1

If the Required Actions and associated Completion Time of Condition A are not met, or if both CVS makeup isolation valves are not OPERABLE (i.e., not able to be closed automatically), then the makeup flow path to the RCS must be isolated. Isolation can be accomplished by manually closing the CVS makeup isolation MOVs or alternatively, manual valve(s) in the makeup line between the makeup pumps and the RCS.

The Action is modified by a Note allowing the flow path 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 main control room. In this way, the flow path can be rapidly isolated when a need for isolation is indicated.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

Verification that the RCS makeup isolation valves are OPERABLE, by stroking each valve closed, demonstrates that the valves can perform their safety related function. The Frequency is in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.17.2

Verification that the RCS makeup isolation valves closure times are less than that assumed in the safety analysis, is performed by measuring the time required for each valve to close. The Frequency is in accordance with the Inservice Testing Program.

REFERENCES

1. Chapter 15, “Accident Analysis.”

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 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."

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 5.5.4, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.4, 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 5.5.4. 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 3.4.7, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via 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 includes primary to secondary SG tube LEAKAGE equivalent to the operational leakage limit of 150 gpd per SG. 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 3.4.10, "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 50.34 (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(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair 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 5.5.4, "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.

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 150 gpd 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 3.4.7, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one 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.

A.1 and A.2

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 3.4.18.2. 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 Times 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 3.4.18.1

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 repair 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 3.4.18.1. 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 5.5.4 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

<u>SR 3.4.18.2</u>

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.5.4 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 repair 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 repair 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 50.34.

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 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the PXS accumulators are to supply water to the reactor vessel during the blowdown phase of a large-break loss-of-coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, to provide Reactor Coolant System (RCS) makeup for a small-break LOCA, and to provide RCS boration for steam line breaks (Ref. 2).

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 the RCS pressure falls to a value approaching that of the containment atmosphere.

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 accumulator inventory is 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.

The accumulators are pressure vessels, partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required for them to perform their function. Internal accumulator pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the static accumulator pressure.

Each accumulator is piped into the reactor vessel via an accumulator line and is isolated from the RCS by two check valves in series.

A normally open motor operated valve is arranged in series with the check valves. Upon initiation of a safeguards actuation signal, the normally open valves receive a confirmatory open signal.

Power lockout and position alarms ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 603-1991 (Ref. 1) for “operating bypasses” and that the accumulators will be available for injection without being subject to a single failure.

BASES

BACKGROUND (continued)

The accumulator size, water volume, and nitrogen cover pressure are selected so that both of the accumulators are sufficient to recover the core cooling before significant clad melting or zirconium water reaction can occur following a large break LOCA. One accumulator is adequate during a small break LOCA where the entire contents of one accumulator can possibly be lost via the pipe break. This accumulator performance is based on design basis accident (DBA) assumptions and models (Ref. 3). The Probabilistic Risk Assessment (PRA) (Ref. 4) shows that one of the two accumulators is sufficient for a large break LOCA caused by spurious ADS actuation and that none of the accumulators are required for small break LOCAs, assuming that at least one core makeup tank (CMT) is available. In addition, both accumulators are required for a large break LOCA caused by the break of a cold leg pipe; the probability of this break has been significantly reduced by incorporation of leak-before-break.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed to be OPERABLE in both the large and small break LOCA analyses at full power (Ref. 3) 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.

For a small break LOCA, a large range of break sizes and locations were analyzed to verify the adequacy of the design. The cases analyzed include the rupture of one 8 inch direct vessel injection line and several smaller break sizes. Acceptable PXS performance was demonstrated.

For a larger LOCA, including a double ended RCS piping rupture, the PXS can provide a sufficiently large flow rate, assuming both accumulators are OPERABLE, to quickly fill the reactor vessel lower plenum and downcomer. Both accumulators, in conjunction with the CMTs, ensure rapid reflooding of the core. For a large LOCA, both lines are available since an 8 inch line break would be a small LOCA.

Following a non-LOCA event such as a steamline break, the RCS experiences a decrease in temperature and pressure due to an increase in energy removal by the secondary system. The cooldown results in a reduction of the core SHUTDOWN MARGIN with a potential for return to power. During such an event the accumulators provide injection of borated water to assist the CMT's boration to mitigate the reactivity transient and ensure the core remains shut down.

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

This LCO establishes the minimum conditions necessary to ensure that sufficient accumulator flow will be available to meet the necessary acceptance criteria established for core cooling by 10 CFR 50.46 (Ref. 5). These conditions are:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

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. The 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 an accumulator to be OPERABLE, the isolation valve must be fully open with power removed, and the limits established in the Surveillance Requirements for contained water, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODES 3 and 4 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 adequate injection flow from other sources exists to retain peak clad temperatures below the 10 CFR 50.46 limit of 2200°F.

In MODES 3 and 4 with RCS pressure ≤ 1000 psig, and in MODES 5 and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows the RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

BASES

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, action must be taken to restore the parameter.

Deviations in boron concentration are expected to be slight, considering that the pressure and volume are verified once per 12 hours. For one accumulator, boron concentration not within limits will have an insignificant effect on the ability of the accumulators to perform their safety function. Therefore, a Completion Time of 72 hours is considered to be acceptable.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 8 hours. With one accumulator inoperable, the remaining accumulator is capable of providing the required safety function, except for one low probability event (large cold leg LOCA) discussed in the background section. The effectiveness of one accumulator is demonstrated in analysis performed to justify PRA success criteria (Ref. 4). The analysis contained in this reference shows that for a range of other events including small LOCAs and large hot leg LOCAs that with one accumulator unavailable the core is adequately cooled. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk (Ref. 7).

The 8 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 is reasonable since the CMTs are required to be available to provide small break LOCA mitigation (i.e., entry into Condition C or E of LCO 3.5.2 has not occurred). The effectiveness of backup CMT injection is demonstrated in analysis performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a small LOCA, the injection from one CMT without any accumulator injection supports adequate core cooling. This analysis provides a high confidence that with the unavailability of one accumulator, the core can be cooled following design bases accidents.

The 1 hour Completion Time, in the case with simultaneous entry into Condition C or E of LCO 3.5.2, requires very prompt actions to restore either the accumulator or the CMT to OPERABLE status. This Completion Time is considered reasonable because of the low probability of simultaneously entering these multiple PXS Conditions and the very small likelihood of a LOCA occurring at the same time.

ACTIONS (continued)

C.1 and C.2

If the Required Action and associated Completion Time of Conditions A or B are not met, the plant must be placed in a MODE or condition in which the LCO does not apply. This is done by placing the plant in MODE 3 within 6 hours and with pressurizer pressure 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.

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 each accumulator isolation valve is fully open, as indicated in the control room, and 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 partially closed valve could result in not meeting DBA analyses assumptions (Ref. 3). A 12 hour Frequency is considered reasonable in view of the other administrative controls which ensure that a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and 3.5.1.3

Verification every 12 hours of the borated water volume and nitrogen cover pressure in each accumulator 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. Considering that control room alarms are provided for both parameters these limits are effectively subject to continuous monitoring. The 12 hour Frequency is considered reasonable considering the availability of the control room alarms and the likelihood that, with any deviation which may occur, the accumulators will perform their safety function with slight deviations in these parameters.

SURVEILLANCE REQUIREMENTS (continued)

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 in-leakage. Sampling the affected accumulator within 6 hours after a 3% volume increase will promptly identify whether the volume change has caused a reduction of boron 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. 6).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the pressurizer 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, reduced accumulator capacity might be available for injection following a DBA that required operation of the accumulators. 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 pressurizer pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startup or shutdowns.

Should closure of a valve occur, the safeguard actuation signal provided to the valve would open a closed valve, if required.

SR 3.5.1.6

This SR requires performance of a system performance test of each accumulator to verify flow capabilities. The system performance test demonstrates that the accumulator injection line resistance assumed in accident analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

BASES

REFERENCES

1. IEEE Standard 603-1991, “Criteria for Safety Systems for Nuclear Power Generating Stations.”

2. Section 6.3 “Passive Core Cooling System.”

3. Section 15.6 “Decrease in Reactor Coolant Inventory.”

4. AP1000 PRA.

5. 10 CFR 50.46.

6. NUREG-1366, February 1990.

7. Regulatory Guide 1.177, 8/98, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications.”

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.2 Core Makeup Tanks (CMTs) – Operating

BASES

BACKGROUND

Two redundant CMTs provide sufficient borated water to assure Reactor Coolant System (RCS) reactivity and inventory control for all design basis accidents (DBAs), including both loss of coolant accident (LOCA) events and non-LOCA events (Ref. 1).

The CMTs are cylindrical tanks with hemispherical upper and lower heads. They are made of carbon steel and clad on the internal surfaces with stainless steel. They are located in containment at an elevation slightly above the reactor coolant loops. Each tank is full of borated water at > 3400 ppm. During normal operation, the CMTs are maintained at RCS pressure through a normally open pressure balance line from the cold leg.

The outlet line from each CMT is connected to one of two direct vessel injection lines, which provides an injection path for the water supplied by the CMT. The outlet line from each CMT is isolated by parallel, normally closed, fail open valves. Upon receipt of a safeguards actuation signal, these four valves open to align the CMTs to the RCS.

The CMTs will inject to the RCS as inventory is lost and steam or reactor coolant is supplied to the CMT to displace the water that is injected. Steam or reactor coolant is provided to the CMT through the cold leg balance line, depending upon the specific event that has occurred. The inlet line from the cold leg is sized for LOCA events, where the cold legs become voided and higher CMT injection flows are required.

The injection line from each CMT contains a flow tuning orifice that is used to provide a mechanism for the field adjustment of the injection line resistance. The orifice is used to establish the required flow rates for the associated plant conditions assumed in the CMT design. The CMT flow is based on providing injection for a minimum of 20 minutes after CMT actuation.

The CMT size and injection capability are selected to provide adequate RCS boration and safety injection for the limiting DBA. One CMT is adequate for this function during a small break LOCA where one CMT completely spills via the pipe break (Ref. 2). The Probabilistic Risk Assessment (PRA) (Ref. 3) shows that none of the CMTs are required for small LOCAs, assuming that at least one accumulator is available.

BASES

APPLICABLE SAFETY ANALYSES

The CMTs are assumed to be OPERABLE to provide emergency boration and core makeup when the Chemical and Volume Control System (CVS) is inoperable, and to mitigate the consequences of any DBA which requires the safety injection of borated water (Ref. 2).

Following a non-LOCA event such as a steamline break, the RCS experiences a decrease in temperature and pressure due to an increase in energy removal by the secondary system. The cooldown results in a reduction of the core SHUTDOWN MARGIN due to the negative moderator temperature coefficient, with a potential for return to power. The actuation of the CMTs following this event provides injection of borated water to mitigate the reactivity transient and ensure the core remains shut down.

In the case of a steam generator tube rupture (SGTR), CMT injection provides borated water to compensate for RCS LEAKAGE.

In the case of an RCS leak of 10 gallons per minute, the CMTs can delay depressurization for at least 10 hours, providing makeup to the RCS and remain able to provide the borated water to compensate for RCS shrinkage and to assure the RCS boration for a safe shutdown.

In the case of a LOCA, the CMTs provide a relatively large makeup flow rate for approximately 20 minutes, in conjunction with the accumulators to provide the initial core cooling.

CMTs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO establishes the minimum conditions necessary to ensure that sufficient CMT flow will be available to meet the initial conditions assumed in the safety analyses. The volume of each CMT represents 100% of the total injected flow assumed in LOCA analysis. If the injection line from a single CMT to the vessel breaks, no single active failure on the other CMT will prevent the injection of borated water into the vessel. Thus the assumptions of the LOCA analysis will be satisfied.

For non-LOCA analysis, two CMTs are assumed. Note that for non-LOCA analysis, the accident cannot disable a CMT.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4 when the RCS is not being cooled by the Normal Residual Heat Removal System (RNS) the CMTs are required to be OPERABLE to provide borated water for RCS inventory makeup and reactivity control following a design basis event and subsequent cooldown.

The CMT requirements in MODE 5 with the RCS pressure boundary intact are specified in LCO 3.5.3, “Core Makeup Tanks (CMTs) – Shutdown, RCS Intact.”

The CMTs are not required to be OPERABLE while in MODE 5 with the RCS pressure boundary open or in MODE 6 because the RCS is depressurized and borated water can be supplied from the In-containment Refueling Water Storage Tank (IRWST), if needed.

In the unlikely event of a total loss of AC power sources, coupled with an inoperable Passive Residual Heat Removal Heat Exchanger (PRHR HX) (beyond DBA), the CMTs may be used in a feed and bleed sequence to remove heat from the RCS.

ACTIONS

A.1

With one outlet isolation valve inoperable on one CMT, action must be taken to restore the valve. In this Condition, the CMT is capable of performing its safety function, provided a single failure of the remaining parallel isolation valve does not occur. A Completion Time of 72 hours is acceptable for two train ECCS systems which are capable of performing their safety function without a single failure.

B.1

If the water temperature or boron concentration of one CMT is not within limits, it must be returned to within limits within 72 hours. The deviations in these parameters are expected to be slight, considering the frequent surveillances and control room monitors. With the temperature above the limit, the full core cooling capability assumed in the safety analysis may not be available. With the boron concentration not within limits, the ability to maintain subcriticality following a DBA may be degraded. However, because only one of two CMTs is inoperable, and the deviations of these parameters are expected to be slight, it is probable that more than a required amount of boron and cooling capability will be available to meet the conditions assumed in the safety analysis.

BASES

ACTIONS (continued)

Since the CMTs are redundant, safety class components, the 72 hour Completion Time is consistent with the times normally allowed for this type of component.

C.1

With two CMTs inoperable due to water temperature or boron concentration, at least one CMT must be restored to within limits in 8 hours. The deviations in these parameters are expected to be slight, considering the frequent surveillances and control room monitors. A Completion Time of 8 hours is considered reasonable since the CMTs are expected to be capable of performing their safety function with slight deviations in these parameters and the accumulators are required to be available for LOCA mitigation (i.e., entry into Condition B of LCO 3.5.1 has not occurred). The effectiveness of accumulator injection is demonstrated in analysis performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a small LOCA, the injection from one accumulator without any CMT injection supports adequate core cooling. This analysis provides a high confidence that with the unavailability of two CMTs due to water temperature or boron concentration deviations, the core can be cooled following design bases accidents.

The 1 hour Completion Time, in the case with simultaneous entry into Condition B of LCO 3.5.1, requires very prompt actions to restore either the CMT or the accumulator to OPERABLE status. This Completion Time is considered reasonable because of the low probability of simultaneously entering these multiple PXS Conditions and the very small likelihood of a LOCA occurring at the same time.

D.1

Excessive amounts of noncondensible gases in a CMT inlet line may interfere with the natural circulation flow (hot water from the RCS through the balance line into the CMT and cold water from the CMT through the direct vessel injection line into the vessel) assumed in the safety analyses for some transients. For CMT injection following a LOCA (steam will enter the CMT through the balance line, displacing the CMT water), gases in the CMT inlet line are not detrimental to the CMT function. The presence of some noncondensible gases does not mean that the CMT natural circulation capability is immediately inoperable, but that gases are collecting and should be vented. The venting of these gases requires containment entry to manually operate the vent valves. A Completion Time of 24 hours is permitted for venting noncondensible gases and is

ACTIONS (continued)

acceptable, since, for the transients, the natural circulation capability of one CMT is adequate to ensure mitigation assuming less conservative analysis assumptions regarding stuck rods and core characteristics.

E.1

With one CMT inoperable for reasons other than Condition A, B, C, D, operation of the CMT may not be available. Action must be taken to restore the inoperable CMT to OPERABLE status within 8 hours. The remaining CMT is sufficient for DBAs except for LOCA in the OPERABLE CMTs DVI line. The 8 hour Completion Time is based on the required availability of injection from the accumulators (provided that entry into Condition B of LCO 3.5.1 has not occurred) to provide SI injection. The effectiveness of accumulator injection is demonstrated in analysis performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a small LOCA, the injection from one accumulator without any CMT supports adequate core cooling. This analysis provides a high confidence that with the unavailability of one CMT, the core can be cooled following design bases accidents.

The 1 hour Completion Time, in the case with simultaneous entry into Condition B of LCO 3.5.1, requires very prompt actions to restore either the CMT or the accumulator to OPERABLE status. This Completion Time is considered reasonable because of the low probability of simultaneously entering these multiple PXS Conditions and the very small likelihood of a LOCA occurring at the same time.

F.1 and F.2

If the Required Action or associated Completion Time of Condition A, B, C, D, or E are not met or the LCO is not met for reasons other than Conditions A through 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 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 3.5.2.1 and SR 3.5.2.2

Verification every 24 hours and 7 days that the temperature and the volume, respectively, of the borated water in each CMT is within limits ensures that when a CMT is needed to inject water into the RCS, the injected water temperature and volume will be within the limits assumed

BASES

SURVEILLANCE REQUIREMENTS (continued)

in the accident analysis. The 24 hour Frequency is adequate, based on the fact that no mechanism exists to rapidly change the temperature of a large tank of water such as a CMT. These parameters are normally monitored in the control room by indication and alarms. Also, there are provisions for monitoring the temperature of the inlet and outlet lines to detect in-leakage which may affect the CMT water temperature.

SR 3.5.2.3

Each CMT inlet isolation valve must be verified to be fully open each 12 hours. Frequent verification is considered to be important, since a CMT can not perform its safety function, if the valve is closed. Control room instrumentation is normally available for this verification.

SR 3.5.2.4

Verification that excessive amounts of noncondensible gases are not present in the inlet line is required every 24 hours. The inlet line of each CMT has a vertical section of pipe which serves as a high point collection point for noncondensible gases. Control room indication of the water level in the high point collection point is available to verify that noncondensible gases have collected to the extent that the water level is depressed below the allowable level. The 24 hour Frequency is based on the expected low rate of gas accumulation and the availability of control room indication.

SR 3.5.2.5

Verification every 7 days that the boron concentration in each CMT is within the required limits ensures that the reactivity control from each CMT, assumed in the safety analysis, will be available as required. The 7 day Frequency is adequate to promptly identify changes which could occur from mechanisms such as in-leakage.

SR 3.5.2.6

Verification that the redundant outlet isolation valves are OPERABLE by stroking the valves open ensures that each CMT will function as designed when these valves are actuated. Prior to opening the outlet isolation valves, the inlet isolation valve should be closed temporarily. Closing the inlet isolation valve ensures that the CMT contents will not be diluted or heated by flow from the RCS. Upon completion of the test, the inlet isolation valves must be opened. The Surveillance Frequency references the inservice testing requirements.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

This SR requires performance of a system performance test of each CMT to verify flow capabilities. The system performance test demonstrates that the CMT injection line resistance assumed in DBA analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

REFERENCES

1. Section 6.3, “Passive Core Cooling System.”
2. Chapter 15, “Accident Analysis.”
3. AP1000 PRA.

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.3 Core Makeup Tanks (CMTs) – Shutdown, RCS Intact

BASES

BACKGROUND	A description of the CMTs is provided in the Bases for LCO 3.5.2, “Core Makeup Tanks – Operating.”
APPLICABLE SAFETY ANALYSES	When the plant is shutdown with the Reactor Coolant System (RCS) pressure boundary intact, the CMT and Passive Residual Heat Removal (PRHR) are the preferred methods for mitigation of postulated events such as loss of normal decay heat removal capability (either loss of Startup Feedwater or loss of normal residual heat removal system). The CMT and PRHR are preferred because the RCS pressure boundary can remain intact, thus preserving one of the barriers to fission product release. For these events, the PRHR provides the safety related heat removal path. And the CMT maintains RCS inventory control. These events can also be mitigated by In-containment Refueling Water Storage Tank (IRWST) injection; however, the RCS must be depressurized (vented) in order to facilitate IRWST injection. Since no loss of coolant accidents (LOCAs) are postulated during MODES 5 and 6, the possibility of a break in the direct vessel injection line is not considered. As a result, only one CMT is required to be available to provide core cooling in response to postulated events. The two parallel CMT outlet isolation valves ensure that injection from one CMT occurs in the event of a single active failure. CMTs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO establishes the minimum conditions necessary to ensure that one CMT will be available for RCS inventory control in the event of the loss of normal decay heat removal capability. The two CMT outlet isolation valves must be OPERABLE to ensure that at least one valve will operate, assuming that the other valve is disabled by a single active failure.
APPLICABILITY	In MODE 4 without steam generator heat removal and in MODE 5 with the RCS pressure boundary intact, one CMT is required to provide borated water to the RCS in the event the nonsafety related chemical and volume control system makeup pumps are not available to provide RCS inventory control.

BASES

APPLICABILITY (continued)

The CMT requirements in MODES 1, 2, 3, and 4 are specified in LCO 3.5.2, “Core Makeup Tanks (CMTs) – Operating.”

The CMTs are not required to be OPERABLE while in MODE 5 with the RCS open or in MODE 6 because the RCS is depressurized and borated water can be supplied from the IRWST, if needed.

ACTIONS

A.1

With one outlet isolation valve inoperable action must be taken to restore the valve. In this Condition the CMT is capable of performing its safety function, provided a single failure of the remaining parallel isolation valve does not occur. A Completion Time of 72 hours is consistent with times normally applied to an ECCS system which is capable of performing its safety function without a single failure.

B.1

If the water temperature or boron concentration in the CMT is not within limits, it must be returned to within limits within 72 hours. With the temperature above the limit the makeup capability assumed in the safety analysis may not be available. With the boron concentration not within limits, the ability to maintain subcriticality may be degraded.

Because the mechanisms for significantly altering these parameters in the CMT are limited, it is probable that more than the required amount of boron and cooling capacity will be available to meet the conditions assumed in the safety analysis. Therefore, the 72 hour Completion Time is acceptable.

C.1

With the required CMT inoperable for reasons other than Condition A or B operation of the CMT may not be available. Action must be taken to restore the inoperable CMT to OPERABLE status within 8 hours. LOCAs are not postulated during the MODEs when this LCO is applicable. The only safety function is to provide LEAKAGE makeup in case normal RCS makeup is unavailable. The 8 hour Completion Time is based on the availability of injection from the IRWST to provide RCS makeup. The ability of the IRWST to provide RCS injection is demonstrated by analysis performed to show that IRWST injection together with ADS venting provides adequate core cooling. Such analysis was performed for the loss of RNS cooling during midloop operations. The analysis was performed in support of the AP1000 PRA (Ref. 2).

BASES

ACTIONS (continued)

D.1

If the Required Action or associated Completion Time of Conditions A, B, or C are not met or the LCO is not met for reasons other than Conditions A through C, action must be initiated, immediately, to place the plant in a MODE where this LCO does not apply. Action must be initiated, immediately, to place the plant in MODE 5 with RCS pressure boundary open and ≥ 20% pressurizer level. In this condition, core cooling and RCS makeup are provided by IRWST injection and sump recirculation. Opening of the ADS valves ensures that IRWST injection can occur.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The LCO 3.5.2 Surveillance Requirements (SR 3.5.2.1 through 3.5.2.7) are applicable to the CMT required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.5.2 for a discussion of each SR.

REFERENCES

1. Section 6.3, “Passive Core Cooling System.”
2. AP1000 PRA.

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.4 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating

BASES

BACKGROUND

The normal heat removal mechanism is the steam generators, which are supplied by the startup feedwater system. However, this path utilizes non-safety related components and systems, so its failure must be considered. In the event the steam generators are not available to remove decay heat for any reason, including loss of startup feedwater, the heat removal path is the PRHR HX (Ref. 1).

The principle component of the PRHR HX is a 100% capacity heat exchanger mounted in the In-containment Refueling Water Storage Tank (IRWST). The heat exchanger is connected to the Reactor Coolant System (RCS) by a inlet line from one RCS hot leg, and an outlet line to the associated steam generator cold leg channel head. The inlet line to the passive heat exchanger contains a normally open, motor operated isolation valve. The outlet line is isolated by two parallel, normally closed air operated valves, which fail open on loss of air pressure or control signal. There is a vertical collection point at the top of the common inlet piping high point which serves as a gas collector. It is provided with level detectors that indicate when noncondensible gases have collected in this area. There are provisions to manually vent these gases to the IRWST.

In order to preserve the IRWST water for long term PRHR HX operation, a gutter is provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation any water collected by the gutter is directed to the normal containment sump. During PRHR HX operation, redundant series air operated valves are actuated to block the draining of condensate to the normal sump and to force the condensate into the IRWST. These valves fail closed on loss of air pressure or control signal.

The PRHR HX size and heat removal capability is selected to provide adequate core cooling for the limiting non-LOCA heatup Design Basis Accidents (DBAs) (Ref. 2). The Probability Risk Assessment (PRA) (Ref. 3) shows that PRHR HX is not required assuming that passive feed and bleed is available. Passive feed and bleed uses the Automatic Depressurization System (ADS) for bleed and the CMTs/accumulators/ IRWST for feed.

BASES

APPLICABLE SAFETY ANALYSES	In the event of a non-LOCA DBA during normal operation, the PRHR HX is automatically actuated to provide decay heat removal path in the event the normal path through the steam generators is not available (Ref. 2). The non-LOCA events which establish the PRHR HX parameters are those involving a decrease in heat removal by the secondary system, such as loss of main feedwater or other failure in the feedwater system. Since the PRHR HX is passive, it will mitigate the consequences of these events with a complete loss of all AC power sources. The PRHR HX actuates when the CMTs are actuated during LOCA events. The PRHR HX satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO requires that the PRHR HX be OPERABLE so that it can respond appropriately to the DBAs which may require its operation. Since this is a passive component, it does not require the actuation of active components such as pumps for its OPERABILITY and will be OPERABLE if the inlet valves are in their normally open position, and the normally closed, fail open outlet valves open on receipt of an actuation signal. In addition to the appropriate valve configuration, OPERABILITY may be impaired by flow blockage caused by noncondensible gases collecting in the system. Thus the absence of noncondensible gases in the high point is necessary for system OPERABILITY. The note requires a reactor coolant pump (RCP) to be operating in the loop with the PRHR HX, Loop 1, if any RCPs are operating. If RCPs are only operating in Loop 2 and no RCPs are operating in Loop 1, there is a possibility there may be reverse flow in the PRHR HX.
APPLICABILITY	The PRHR HX must be OPERABLE in MODES 1, 2, 3, and 4 with the RCS not cooled by the Normal Residual Heat Removal System (RNS) if a plant cooldown is required and the normal cooldown path is not available. Under these conditions, the PRHR HX may be actuated to provide core cooling and to mitigate the consequences of a DBA. The PRHR HX requirements in MODE 4 with RCS cooling provided by the RNS and in MODE 5 with the RCS pressure boundary intact are specified in LCO 3.5.5, “Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Shutdown, RCS Intact.” The PRHR HX is not capable of natural circulation cooling of the RCS in MODE 5 with the RCS pressure boundary open or in MODE 6.

BASES

ACTIONS

A.1

The outlet line from the PRHR HX is controlled by a pair of normally closed, fail open, air operated valves, arranged in parallel. Thus they are redundant and, if either valve is OPERABLE, the system can function at 100% capacity, assuming other OPERABILITY conditions are met.

If one valve is inoperable, a Completion Time of 72 hours has been allowed to restore the inoperable valve(s) to OPERABLE status. This Completion Time is consistent with the Completion Times specified for other parallel redundant safety related systems.

B.1

With one air operated IRWST gutter isolation valve inoperable, the remaining isolation valve can function to drain the gutter to the IRWST. Action must be taken to restore the inoperable gutter isolation valve to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable based on the capability of the remaining valve to perform 100% of the required safety function assumed in the safety analyses.

C.1

Excessive amounts of noncondensible gases in the PRHR HX inlet line may interfere with the natural circulation flow of reactor coolant through the PRHR HX. The presence of some noncondensible gases does not mean that the PRHR HX is immediately inoperable, but that gases are collecting and should be vented. The venting of these gases requires containment entry to manually operate the appropriate vent valves. A Completion Time of 24 hours is acceptable considering that passive feed and bleed cooling is available to remove heat from the RCS.

D.1 and D.2

If any of the above Required Actions have not been accomplished in the required Completion Time or the LCO is not met for reasons other than Conditions A, B, or C, 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, with the RCS cooled by the RNS, 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.

BASES

ACTIONS (continued)

E.1

With the LCO not met for reasons other than Condition A, B, or C, the PRHR HX must be restored within 8 hours. The 8 hour Completion Time is based on the availability of passive feed and bleed cooling to provide RCS heat removal. The effectiveness of feed and bleed cooling has been demonstrated in analysis and evaluations performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a range of events including loss of main feedwater, SGTR, and small LOCA (as small as 1/2") that feed and bleed cooling provides adequate core cooling.

These analyses and evaluations provide a high confidence that with the unavailability of the PRHR HX the core can be cooled following design bases accidents.

F.1 and F.2

If the PRHR HX is not restored in accordance with Action E.1 within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 5 within 36 hours.

Action F.1 is modified by a Note which requires that prior to initiating cooldown of the plant to MODE 3, redundant means of providing SG feedwater be verified as OPERABLE. Possible means include main feedwater and startup feedwater pumps. With the PRHR HX and redundant means of feeding the SGs INOPERABLE, the unit is in a seriously degraded condition with no means for conducting a controlled cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. If redundant means of feeding the SGs are not available, the plant should be maintained in the current MODE until redundant means are restored. LCO 3.0.3 and all other Required Actions shall be suspended until the redundant means are restored, because they could force the unit into a less safe condition.

Action F.2 is modified by a Note which requires that prior to stopping SG feedwater, redundant means of cooling the RCS to cold shutdown conditions must be verified as OPERABLE. One redundant means of cooling the RCS to cold shutdown includes the normal residual heat removal system (RNS) and its necessary support system (both component cooling system pumps and heat exchangers, and both service water system pumps and fans). Without availability of these redundant

BASES

ACTIONS (continued)

cooling means, the unit is in a seriously degraded condition with no means for continuing the controlled cooldown. Until the redundant cooling means are restored, heat removal using SG feedwater should be maintained. LCO 3.0.3 and all other Required Actions shall be suspended until the systems and equipment required for further cooldown are restored, because they could force the unit into a less safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

Verification, using remote indication, that the common outlet manual isolation valve is fully open ensures that the flow path from the heat exchangers to the RCS is available. Misalignment of this valve could render the heat exchanger inoperable. A 12 hour Frequency is reasonable considering that the valve is manually positioned and has control room position indication and alarm.

SR 3.5.4.2

Verification that the motor operated inlet valve is fully open, as indicated in the main control room, ensures timely discovery if the valve is not fully open. The 12 hour Frequency is consistent with the ease of verification, confirmatory open signals, and redundant series valve controls that prevent spurious closure.

SR 3.5.4.3

Verification that excessive amounts of noncondensible gases are not present in the inlet line is required every 24 hours. The inlet line of the PRHR HX has a vertical section of pipe which serves as a high point collection point for noncondensible gases. Control room indication of the water level in this high point collection point is available to verify that noncondensible gases have not collected to the extent that the water level is depressed below the allowable level. The 24 hour Frequency is based on the expected low rate of gas accumulation and the availability of control room indication.

SR 3.5.4.4

Verification is required to confirm that power is removed from the motor operated inlet isolation valve every 31 days. Removal of power from this valve reduces the likelihood that the valve will be inadvertently closed as a result of a fire. The 31 day Frequency is acceptable considering the frequent surveillance of valve position and that the valve has a confirmatory open signal.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.5

Verification that both air operated outlet valves and both IRWST gutter isolation valves are OPERABLE ensures that the PRHR HX will actuate on command, with return flow from the gutter to the IRWST, since all other components of the system are normally in the OPERABLE configuration. Since these valves are redundant, if one valve is inoperable, the system can function at 100% capacity. Verification requires the actual operation of each valve through a full cycle to demonstrate OPERABILITY. The Surveillance Frequency is provided in the Inservice Testing Program.

SR 3.5.4.6

This SR requires performance of a system performance test of the PRHR HX to verify system heat transfer capabilities. The system performance test demonstrates that the PRHR HX heat transfer assumed in accident analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

SR 3.5.4.7

This surveillance requires visual inspection of the IRWST gutters to verify that the return flow to the IRWST will not be restricted by debris. A Frequency of 24 months is adequate, since there are no known sources of debris with which the gutters could become restricted.

REFERENCES

1. Section 6.3, “Passive Core Cooling System.”
2. Chapter 15, “Safety Analysis.”
3. AP1000 PRA.

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.5 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Shutdown, RCS Intact

BASES

BACKGROUND	A description of the PRHR HX is provided in the Bases for LCO 3.5.4, “Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating.”
APPLICABLE SAFETY ANALYSES	In the event of a loss of normal decay heat removal capability during shutdown with the Reactor Coolant System (RCS) pressure boundary intact, the PRHR HX provides the preferred safety related heat removal path. When required, the PRHR HX is manually actuated and can maintain the RCS < 420°F. Alternatively, the heat removal function can be provided by depressurizing the RCS with the Automatic Depressurization System (ADS) and injection of the In-containment Refueling Water Storage Tank (IRWST) with containment closure capability provided. The PRHR HX is preferred because the RCS pressure boundary remains intact, thus preserving a barrier to fission product release. The PRHR HX satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO requires the PRHR HX to be OPERABLE so that it can be placed in service in the event normal decay heat removal capability is lost. Since this a passive component, it does not require the actuation of active components such as pumps for its OPERABILITY and will be OPERABLE if the inlet valves are in their normally open position, and the normally closed, fail open outlet valves open on receipt of an actuation signal. In addition to the appropriate valve configuration, OPERABILITY may be impaired by flow blockage caused by noncondensible gases collecting in the system. Thus the absence of non-condensible gases in the high point is necessary for system OPERABILITY. The note requires a reactor coolant pump (RCP) to be operating in the loop with the PRHR HX, Loop 1, if any RCPs are operating. If RCPs are only operating in loop 2 and no RCPs are operating in loop 1, there is a possibility there may be reverse flow in the PRHR HX.

APPLICABILITY

The PRHR HX must be OPERABLE in MODE 4 with RCS cooling provided by the RNS and in MODE 5 with the RCS pressure boundary intact and pressurizer level ≥ 20% to provide decay heat removal in the event the normal residual heat removal system is not available.

The PRHR HX requirements in MODES 1, 2, 3, and 4 with RCS cooling not provided by the RNS are specified in LCO 3.5.4, "Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating."

The PRHR HX is not capable of natural circulation cooling of the RCS in MODE 5 with either the RCS pressure boundary open or with the RCS intact when pressurizer level ≤ 20%, or in MODE 6.

ACTIONS

A.1

The outlet line from the PRHR HX is isolated by a pair of normally closed, fail open, air operated valves, arranged in parallel. They are redundant, and if either valve is OPERABLE the system can function at 100% capacity, assuming other OPERABILITY conditions are met.

Since these valves are redundant, if one valve is inoperable, a Completion Time of 72 hours has been allowed to restore the inoperable valve to OPERABLE status. This Completion Time is consistent with the Completion Times specified for other parallel redundant safety related systems.

B.1

With one air operated IRWST gutter isolation valve inoperable, the remaining isolation valve can function to drain the gutter to the IRWST. Action must be taken to restore the inoperable gutter isolation valve to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable based on the capability of the remaining valve to perform 100% of the required safety function assumed in the safety analyses.

C.1

At the inlet piping high point there is a vertical chamber which serves as a collection point for noncondensible gases. This collection point is provided with detectors which alarm to indicate when gases have collected in this area. The presence of an alarm does not mean that PRHR HX is immediately inoperable, but that gases are collecting and should be vented. A Completion Time of 24 hours is acceptable, considering that passive feed and bleed cooling is available to revise heat from the RCS.

ACTIONS (continued)

D.1

With the LCO not met for reasons other than Condition A, B, or C, the PRHR HX must be restored within 8 hours. The 8 hour Completion Time is acceptable based on the availability of passive feed and bleed cooling to provide RCS heat removal. The effectiveness of feed and bleed cooling is discussed in the bases for LCO 3.5.4, Action E.1.

E.1

If any of the above Required Actions have not been accomplished in the required Completion Time, or the LCO is not met for reasons other than Conditions A, B, C, or D, action must be initiated, immediately, to be in MODE 5 with the RCS pressure boundary open and pressurizer level ≥ 20%. The time to RCS boiling is maximized in the event of loss of normal decay heat removal capability, by maintaining a visible level in the pressurizer. Additionally, in this MODE the RCS must be opened, such that safety related decay heat removal can be immediately initiated by actuation of the IRWST injection valve(s).

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

The LCO 3.5.4 Surveillance Requirements are applicable to the PRHR HX required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.5.4 for a discussion of each SR.

REFERENCES

None.

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.6 In-containment Refueling Water Storage Tank (IRWST) – Operating

BASES

BACKGROUND

The IRWST is a large stainless steel lined tank filled with borated water (Ref. 1). It is located below the operating deck in containment. The tank is designed to meet seismic Category 1 requirements. The floor of the IRWST is elevated above the reactor coolant loop so that borated water can drain by gravity into the Reactor Coolant System (RCS). The IRWST is maintained at ambient containment pressure.

The IRWST has two injection flow paths. The injection paths are connected to the reactor vessel through two direct vessel injection lines which are also used by the accumulators and the core makeup tanks. Each path includes an injection flow path and a containment recirculation flow path. Each injection path includes a normally open motor operated isolation valve and two parallel actuation lines each isolated by one check valve and one squib valve in series.

The IRWST has two containment recirculation flow paths. Each containment recirculation path contains two parallel actuation flow paths, one path is isolated by a normally open motor operated valve in series with a squib valve and one path is isolated by a check valve in series with a squib valve.

During refueling operations, the IRWST is used to flood the refueling cavity. During abnormal events, the IRWST serves as a heat sink for the passive residual heat removal heat exchangers, as a heat sink for the depressurization spargers, and as a source of low head (ambient containment pressure) safety injection during loss of coolant accidents (LOCAs) and loss of decay heat removal in MODE 5 (loops not filled). The IRWST can be cooled by the Normal Residual Heat Removal System (RNS) system.

The IRWST size and injection capability is selected to provide adequate core cooling for the limiting Design Basis Accidents (DBAs) (Ref. 2).

APPLICABLE SAFETY ANALYSES

During non-LOCA events, the IRWST serves as the initial heat sink for the PRHR Heat Exchanger (PRHR HX) if used during reactor cooldown to MODE 4. If RNS is available, it will be actuated in MODE 4 and used to continue the plant cooldown to MODE 5. If RNS is not available, cooldown can continue on PRHR. Continued PRHR HX operation will result in the water in the IRWST heating up to saturation conditions and

BASES

APPLICABLE SAFETY ANALYSES (continued)

boiling. The steam generated in the IRWST enters the containment through the IRWST vents. Most of the steam generated in the IRWST condenses on the inside of the containment vessel and drains back to the IRWST.

For events which involve a loss of primary coolant inventory, such as a large break LOCA, or other events involving automatic depressurization, the IRWST provides low pressure safety injection (Ref. 2). The IRWST drain down time is dependent on several factors, including break size, location, and the return of steam condensate from the passive containment cooling system. During drain down, when the water in the IRWST reaches the Low 5 level, the containment sump will be sufficiently flooded, to initiate containment sump recirculation. This permits continued cooling of the core by recirculation of the spilled water in the containment sumps via the sump recirculation flow paths. In this situation, core cooling can continue indefinitely.

When the plant is in midloop operation, the pressurizer Automatic Depressurization System (ADS) valves are open, and the RNS is used to cool the RCS. The RNS is not a safety related system, so its failure must be considered. In this situation, with the RCS drained and the pressure boundary open, the PRHR HX cannot be used. In such a case, core cooling is provided by gravity injection from the IRWST, venting the RCS through the ADS. Injection from the IRWST provides core cooling until the tank empties and gravity recirculation from the containment starts. With the containment closed, the recirculation can continue indefinitely, with the decay heat generated steam condensing on the containment vessel and draining back into the IRWST.

The IRWST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The IRWST requirements ensure that an adequate supply of borated water is available to act as a heat sink for PRHR and to supply the required volume of borated water as safety injection for core cooling and reactivity control.

To be considered OPERABLE, the IRWST must meet the water volume, boron concentration, and temperature limits defined in the surveillance requirements. The motor operated injection isolation valves must be open with power removed, and the motor operated sump recirculation isolation valves must be open.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a safety related function of the IRWST is to provide a heat sink for PRHR. In MODES 1, 2, 3, 4, and 5, a second safety related function is the low head safety injection of borated water following a LOCA for core cooling and reactivity control. Both of these functions must be available to meet the initial assumptions of the safety analyses. These assumptions require the specified boron concentration, the minimum water volume, and the maximum water temperature.

The requirements for the IRWST in MODES 5 and 6 are specified in LCO 3.5.7, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5 and LCO 3.5.8, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6.

ACTIONS

A.1

If an IRWST injection line actuation valve flow path or a containment recirculation line actuation valve flow path is inoperable, then the valve actuation flow path must be restored to OPERABLE status within 72 hours. In this condition, three other IRWST injection or containment sump recirculation flow paths are available and can provide 100% of the required flow assuming a break in the direct vessel injection line associated with the other injection train, but with no single failure of the actuation valve flow path in the same injection or sump recirculation flow path. The 72 hour Completion Time is consistent with times normally applied to degraded two train ECCS systems which can provide 100% of the required flow without a single failure.

B.1

If the IRWST water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR HX heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The allowable deviation of the water volume is limited to 3%. This limit prevents a significant change in boron concentration and is consistent with the long-term cooling analysis performed to justify PRA success criteria (Ref. 3), which assumed multiple failures with as many as 3 CMTs/Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8-hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to DBAs with slight deviations in these parameters.

BASES

ACTIONS (continued)

C.1

If the motor operated IRWST isolation valves are not fully open or valve power is not removed, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valves must be restored to fully open with valve power removed in 1 hour. This Completion Time is acceptable based on risk considerations.

D.1 and D.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Times or the LCO is not met for reasons other than Conditions A, B, or C, 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 3.5.6.1

The IRWST borated water temperature must be verified every 24 hours to ensure that the temperature is within the limit assumed in the accident analysis. This Frequency is sufficient to identify a temperature change that would approach the limit and has been shown to be acceptable through operating experience.

SR 3.5.6.2

Verification every 24 hours that the IRWST borated water volume is above the required minimum level will ensure that a sufficient initial supply is available for safety injection and floodup volume for recirculation and as the heat sink for PRHR. During shutdown with the refueling cavity flooded with water from the IRWST, this Surveillance requires that the combined volume of borated water in the IRWST and refueling cavity meet the specified limit. Since the IRWST volume is normally stable, and is monitored by redundant main control indication and alarm, a 24 hour Frequency is appropriate.

SR 3.5.6.3

Verification every 31 days that the boron concentration of the IRWST is greater than the required limit, ensures that the reactor will remain subcritical following a LOCA. Since the IRWST volume is large and

SURVEILLANCE REQUIREMENTS (continued)

normally stable, the 31 day Frequency is acceptable, considering additional verifications are required within 6 hours after each solution volume increase of 15,000 gal. In addition, the relatively frequent surveillance of the IRWST water volume provides assurance that the IRWST boron concentration is not changed.

SR 3.5.6.4

This surveillance requires verification that each motor operated isolation valve is fully open. This surveillance may be performed with available remote position indication instrumentation. The 12 hour Frequency is acceptable, considering the redundant remote indication and alarms and that power is removed from the valve operator.

SR 3.5.6.5

Verification is required to confirm that power is removed from each motor operated IRWST isolation valve each 31 days. Removal of power from these valves reduces the likelihood that the valves will be inadvertently closed. The 31 day Frequency is acceptable considering frequent surveillance of valve position and that the valve has a confirmatory open signal.

SR 3.5.6.6

Each motor operated containment recirculation isolation valve must be verified to be fully open. This valve is required to be open to improve containment recirculation reliability. The 31 day Frequency is acceptable considering the valve has a confirmatory open signal. This surveillance may be performed with available remote position indication instrumentation.

SR 3.5.6.7

This Surveillance requires verification that each IRWST injection and each containment recirculation squib valve is OPERABLE in accordance with the Inservice Testing Program. The Surveillance Frequency for verifying valve OPERABILITY references the Inservice Testing Program.

The squib valves will be tested in accordance with the ASME OM Code. The applicable ASME OM Code squib valve requirements are specified in paragraph 4.6, Inservice Tests for Category D Explosively Actuated Valves. The requirements include actuation of a sample of the installed valves each 2 years and periodic replacement of charges.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.6.8

Visual inspection is required each 24 months to verify that the IRWST screens and the containment recirculation screens are not restricted by debris. A Frequency of 24 months is adequate, since there are no known sources of debris with which the gutters could become restricted.

SR 3.5.6.9

This SR requires performance of a system inspection and performance test of the IRWST injection and recirculation flow paths to verify system flow capabilities. The system inspection and performance test demonstrates that the IRWST injection and recirculation capabilities assumed in accident analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

REFERENCES

1. Section 6.3, “Passive Core Cooling.”
2. Section 15.6, “Decrease in Reactor Coolant Inventory.”
3. AP1000 PRA.

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.7 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5

BASES

BACKGROUND

A description of the IRWST is provided in LCO 3.5.6, "In-containment Refueling Water Storage Tank – Operating."

APPLICABLE SAFETY ANALYSES

For postulated shutdown events in MODE 5 with the Reactor Coolant System (RCS) pressure boundary intact, the primary protection is Passive Residual Heat Removal (PRHR), where the IRWST serves as the initial heat sink for the PRHR heat exchanger (PRHR HX). For events in MODE 5 with the RCS pressure boundary open, PRHR is not available and RCS heat removal is provided by IRWST injection and containment sump recirculation.

IRWST injection could be required to mitigate some events by providing RCS inventory makeup.

No loss of coolant accidents (LOCAs) are postulated during plant operation in MODE 5; therefore, the rupture of the direct vessel injection line (DVI) is not assumed. Since the DVI rupture is not assumed, only one train of IRWST injection and recirculation flow paths is required to mitigation postulated events, assuming a single failure.

The IRWST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The IRWST requirements ensure that an adequate supply of borated water is available to act as a heat sink for PRHR and to supply the required volume of borated water as safety injection for core cooling and reactivity control.

To be considered OPERABLE, the IRWST must meet the water volume, boron concentration, and temperature limits defined in the Surveillance Requirements, and one path of injection and recirculation must be OPERABLE (the motor operated injection isolation valve must be open with power removed, and the motor operated sump recirculation isolation valves must be open).

APPLICABILITY

In MODE 5 with the RCS pressure boundary intact or with the RCS open with pressurizer level ≥ 20%, the IRWST is an RCS injection source of borated water for core cooling and reactivity control. Additionally, in MODE 5 with the RCS pressure boundary intact, the IRWST provides the heat sink for PRHR.

APPLICABILITY (continued)

The requirements for the IRWST in MODES 1, 2, 3, and 4 are specified in LCO 3.5.6, In-containment Refueling Water Storage Tank (IRWST) – Operating. The requirements for the IRWST in MODE 6 are specified in LCO 3.5.8, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6.

ACTIONS

A.1

If a motor operated containment sump isolation valve in one sump recirculation flow path is not fully open, the valve must be fully opened within 72 hours. The 72 hour Completion Time is consistent with times normally applied to degraded two train ECCS systems which can provide 100% of the required flow without a single failure.

B.1

If the IRWST water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The allowable deviation of the water volume is limited to 3%. This limit prevents a significant change in boron concentration and is consistent with the long-term cooling analysis performed to justify PRA success criteria (Ref. 3), which assumed multiple failures with as many as 3 CMTs/Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8-hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to DBAs with slight deviations in these parameters.

C.1

If the motor operated IRWST isolation valves are not fully open or valve power is not removed, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valves must be restored to fully open with valve power removed in 1 hour. This Completion Time is acceptable based on risk considerations.

D.1 and D.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Times or the LCO is not met for reasons other than Conditions A, B, or C, the plant must be placed in a condition in which the probability and consequences of an event are minimized to the

ACTIONS (continued)

extent possible. This is done by immediately initiating action to place the plant in MODE 5 with the RCS intact with ≥ 20% pressurizer level. The time to RCS boiling is maximized by maintaining RCS inventory at ≥ 20% pressurizer level and maintaining RCS temperature as low as practical. With the RCS intact, the availability of the PRHR HX is maintained. Additionally, action to suspend positive reactivity additions is required to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

SURVEILLANCE REQUIREMENTS

SR 3.5.7.1

The LCO 3.5.6 Surveillance Requirements and Frequencies (SR 3.5.6.1 through 3.5.6.7) are applicable to the IRWST and the flow paths required to be OPERABLE. Refer to the corresponding Bases for LCO 3.5.6 for a discussion of each SR.

REFERENCES

None.

B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.8 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6

BASES

BACKGROUND	A description of the IRWST is provided in LCO 3.5.6, "In-containment Refueling Water Storage Tank (IRWST) – Operating."
APPLICABLE SAFETY ANALYSES	For MODE 6, heat removal is provided by IRWST injection and containment sump recirculation. IRWST injection could be required to mitigate some events by providing RCS inventory makeup. One line with redundant, parallel valves is required to accommodate a single failure (to open) of an isolation valve. The IRWST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The IRWST requirements ensure that an adequate supply of borated water is available to supply the required volume of borated water as safety injection for core cooling and reactivity control. To be considered OPERABLE, the IRWST in combination with the refueling cavity must meet the water volume, boron concentration, and temperature limits defined in the Surveillance Requirements, and one path of injection and recirculation must be OPERABLE. The motor operated injection isolation valve must be open and power removed, and the motor operated sump recirculation isolation valves must be closed and OPERABLE. Any cavity leakage should be estimated and made up with borated water such that the volume in the IRWST plus the refueling cavity will meet the IRWST volume requirement.
APPLICABILITY	In MODE 6, the IRWST is an RCS injection source of borated water for core cooling and reactivity control. The requirements for the IRWST in MODES 1, 2, 3, and 4 are specified in LCO 3.5.6, In-containment Refueling Water Storage Tank (IRWST) – Operating. The requirements for the IRWST in MODE 5 are specified in LCO 3.5.7, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5.

ACTIONS

A.1

With one motor operated containment sump isolation valve not fully open, the valve must be fully opened within 72 hours. The 72 hour Completion Time is consistent with times normally applied to degraded two train ECCS systems which can provide 100% of the required flow without a single failure.

B.1

If the IRWST and refueling cavity water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR HX heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The allowable deviation of the water volume is limited to 3%. This limit prevents a significant change in boron concentration and is consistent with the long-term cooling analysis performed to justify PRA success criteria (Ref. 3), which assumed multiple failures with as many as 3 CMTs/Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8-hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to DBAs with slight deviations in these parameters.

C.1

If the motor operated IRWST isolation valves are not fully open or valve power is not removed, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valves must be restored to fully open with valve power removed in 1 hour. This Completion Time is acceptable based on risk considerations.

D.1 and D.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Times or the LCO is not met for reasons other than Conditions A, B, C, or D, the plant must be placed in a Condition in which the probability and consequences of an event are minimized to the extent possible. In MODE 6, action must be immediately initiated to be in MODE 6 with the cavity water level ≥ 23 feet above the top of the reactor vessel flange.

The time to RCS boiling is maximized by maximizing the RCS inventory and maintaining RCS temperature as low as practical. With the RCS intact, another means of removing decay heat is available (the PRHR HX). Additionally, action to suspend positive reactivity additions is required to ensure that the SDM is maintained. Sources of positive

ACTIONS (continued)

reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS. These Actions place the plant in a condition which maximizes the time to IRWST injection, thus providing time for repairs or application of alternative cooling capabilities.

SURVEILLANCE REQUIREMENTS

SR 3.5.8.1

The IRWST and refueling cavity borated water temperature must be verified every 24 hours to ensure that the temperature is within the limit assumed in accident analysis. This Frequency is sufficient to identify a temperature change that would approach the limit and has been shown to be acceptable through operating experience.

SR 3.5.8.2

Verification every 24 hours that the IRWST and refueling cavity borated water volume is above the required minimum level will ensure that a sufficient initial supply is available for safety injection and floodup volume for recirculation and as the heat sink for PRHR. During shutdown with the refueling cavity flooded with water from the IRWST, this Surveillance requires that the combined volume of borated water in the IRWST and refueling cavity meet the specified limit. Since the IRWST volume is normally stable, and is monitored by redundant main control indication and alarm, a 24 hour Frequency is appropriate.

SR 3.5.8.3

Verification every 31 days that the boron concentration of the IRWST and refueling cavity is greater than the required limit ensures that the reactor will remain subcritical following shutdown events. Since the IRWST volume is large and normally stable, the 31 day Frequency is acceptable, considering additional verifications are required within 6 hours after each solution volume increase of 15,000 gal.

SR 3.5.8.4

LCO 3.5.6 Surveillance Requirements and Frequencies SR 3.5.6.4 through 3.5.6.8 are applicable to the IRWST and the flow paths required to be OPERABLE. Refer to the corresponding Bases for LCO 3.5.6 for a discussion of each SR.

REFERENCES

None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low-leakage steel vessel designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA) such that offsite radiation exposures are maintained within limits. The containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with elliptical upper and lower heads, completely enclosed by a seismic Category I reinforced concrete shield building. A 4.5 foot wide annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to permit inservice inspection and air flow over the steel dome for containment cooling. The containment utilizes the outer concrete building for shielding and a missile barrier, and the inner steel containment for leak tightness and passive containment cooling.

Containment piping penetration assemblies provide for the passage of process, service and sampling pipelines into the containment vessel while maintaining containment integrity. The shield building provides biological shielding and environmental missile protection for the containment vessel and the Nuclear Steam Supply System.

The inner steel containment 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 Surveillance Requirements conform with 10 CFR 50, Appendix J (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

BASES

BACKGROUND (continued)

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”;

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 DBA without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a 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. The DBA analyses assume 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 is designed with an allowable leakage rate of 0.10% of containment air weight of the original content of containment air after a DBA 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 (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on containment leakage rate testing. L_a is assumed to be 0.10% per day in the safety analysis.

Satisfactory leakage rate test results is a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, 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.

BASES

LCO (continued)

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.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. 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 L_a.

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. The MODES 5 and 6 requirements are specified in LCO 3.6.8, “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. Failure to meet air lock leakage limits specified in LCO 3.6.2 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 leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage, and $< 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ 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.

REFERENCES

1. 10 CFR 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.”
2. Chapter 15, “Accident Analysis.”
3. Section 6.2, “Containment Systems.”

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.

Each air lock is nominally a right circular cylinder, 10 feet in diameter, 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).

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness are 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 DBA that results in the largest release of radioactive material within containment is a loss of coolant accident (LOCA) (Ref. 3). In the analyses of DBAs, 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 is designed with an allowable leakage rate of 0.10% of containment air weight of the original content of containment air per day after a DBA (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a following a DBA. 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(c)(2)(ii).

BASES

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of offsite radiation exposures 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 necessary to support containment OPERABILITY following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry and 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 and large inventory of coolant. Therefore, containment air locks are not required to be OPERABLE in MODES 5 and 6 to prevent leakage of radioactive material from containment. However, containment closure capability is required within MODES 5 and 6 as specified in LCO 3.6.8.

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 to repair without interrupting containment integrity. If containment entry is required, 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 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.

BASES

ACTIONS (continued)

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 that 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 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, "Containment," 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.

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 reasonable based on engineering judgement 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 to be locked closed by 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.

ACTIONS (continued)

The Required Actions are 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 airlock are inoperable. With both doors in the same airlock 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 an airlock for entry and exit for 7 days, under administrative controls if both airlocks 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 Specification (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 (non-TS-related activities) if the containment is entered, using the inoperable airlock, 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 in which the OPERABLE door is expected to be open.

B.1, B.2, and B.3

With an air lock door interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with 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 airlock are inoperable. With both doors in the same airlock 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 airlock to ensure that only one door is opened at a time (the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to airlock doors located in high radiation areas that allows these doors to be verified locked closed by 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.

BASES

ACTIONS (continued)

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.

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 respect to air lock leakage (Type B leakage tests). The acceptance

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 as 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 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 door 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 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 inner and outer door opening 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 month 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. Operating experience has shown these components usually pass the Surveillance when performed at 24 month Frequency. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the airlock.

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Performance-Based Requirements."
2. Section 6.2, "Containment Systems."
3. Chapter 15, "Accident Analysis."

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. 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. Section 6.2 (Ref. 1) identifies parameters which initiate isolation signal generation for containment isolation valves. 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 analysis. Therefore, the OPERABILITY requirements provide assurance that containment function assumed in the safety analysis will be maintained.

<u>Containment Air Filtration System 16-inch purge valves</u>

The Containment Air Filtration System operates to:

a. Supply outside air into the containment for ventilation and cooling or heating,

b. Reduce the concentration of noble gases within containment prior to and during personnel access, and

c. Equalize internal and external pressures.

BASES

BACKGROUND (continued)

Since the valves used in the Containment Air Filtration 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 analysis 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) and a rod ejection accident (Ref. 2). In the analyses for each of the 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 DBA dose analysis assumes that, following containment isolation signal generation, the containment purge isolation valves are closed within 10 seconds. The remainder of the automatic isolation valves are assumed closed and the containment leakage is terminated except for the design leakage rate, L_a. Since the containment isolation valves are powered from the 1E division batteries no diesel generator startup time is applied.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the design of the containment purge isolation 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 on each line are pneumatically operated, spring closed valves that fail in the closed position and are provided with power via independent sources.

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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.

BASES

LCO (continued)

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The valves covered by this LCO are listed along with their associated stroke times in the Section 6.2 (Ref. 1).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, or blind flanges are in place and closed systems are intact. These passive isolation valves/devices are those listed in Reference 1.

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, containment isolation valves are not required to be OPERABLE in MODES 5 and 6 to prevent leakage of radioactive material from containment. However, containment closure capability is required in MODES 5 and 6. The requirements for containment isolation valves during MODES 5 and 6 are addressed in LCO 3.6.8, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing containment penetration flow paths to be unisolated intermittently under administrative control. 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.

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.

ACTIONS (continued)

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.

In the event that the containment 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 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, or 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, the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4, and the availability of a second barrier.

For affected penetrations that cannot be restored to OPERABLE status within the 4 hour Completion Time and have been isolated in accordance with Required Action A.1, the affected penetrations must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations that are required to be isolated following an accident and that 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, through a system walkdown, that those isolation devices outside containment and capable of potentially 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

ACTIONS (continued)

the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

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 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 valves once they have been verified to be in the proper position, is small.

<u>B.1</u>

With two containment isolation valves in one or more penetration flow paths inoperable, 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 deactivated 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 ensure 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.

ACTIONS (continued)

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two 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 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 deactivated 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 that the affected penetration is isolated in accordance with Required Action C.1, the affected penetration 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 penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Ref. 4. 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 controls. Allowing verification by administrative controls 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

BASES

ACTIONS (continued)

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 and D.2

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.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

This SR ensures that the 16 inch purge valves are closed as required or, if open, open for an allowable reason. If a 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 16 inch 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 16 inch 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 3.6.3.2.

SR 3.6.3.2

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, through a system walkdown, that those valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for valves outside containment is relatively easy, the 31 day Frequency is based on

SURVEILLANCE REQUIREMENTS (continued)

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.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by 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.

<u>SR 3.6.3.3</u>

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 specified as “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 control 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 controls. Allowing verification by administrative controls 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.4

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 that the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation times are specified in Section 6.2.3 (Ref. 1) and Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.3.5

Automatic containment isolation valves close on 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. Section 6.2, “Containment Systems.”
2. Chapter 15, “Accident Analysis.”
3. NUREG-1449, “Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States.”
4. Standard Review Plan 6.2.4.

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 steam line break (SLB). 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 operating band of conditions used in the containment pressure analyses for the Design Basis Events which result in internal or external pressure loads on the containment vessel. Should operation occur outside these limits, the initial containment pressure would be outside the range used for containment pressure analyses.

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 SLB, which are analyzed using computer pressure transients (Ref. 1).

The initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig). This resulted in a maximum peak pressure from a LOCA, P_a, of 57.8 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure results from the SLB. The maximum containment pressure resulting from the SLB, 57.3 psig, does not exceed the containment design pressure, 59 psig.

The containment was also designed for an external pressure load equivalent to 2.9 psig. The limiting negative pressure transient is a loss of all AC power sources coincident with extreme cold weather conditions which cool the external surface of the containment vessel. The initial pressure condition used in this analysis was -0.2 psig. This resulted in a minimum pressure inside containment, as illustrated in Reference 1, which is less than the design load. Other external pressure load events evaluated include:

Failed fan cooler control

Malfunction of containment purge system

APPLICABLE SAFETY ANALYSES (continued)

Inadvertent Incontainment Refueling Water Storage Tank (IRWST) drain

Inadvertent Passive Containment Cooling System (PCS) actuation

Since the containment external pressure design limits can be met by ensuring compliance with the initial pressure condition, NUREG-1431 LCO 3.6.12, Vacuum Relief System is not applicable to the AP1000 containment.

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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

A.1

When containment pressure is not within the limits of the LCO, it must be restored 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.

BASES

ACTIONS (continued)

B.1 and B.2

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.

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 both containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the main control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. Section 6.2, “Containment Analysis.”

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 steam line break (SLB).

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 plant operations. The total amount of energy to be removed from containment by the passive containment cooling system 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 SLB. The DBA LOCA and SLB 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 containment Engineered Safety Feature (ESF) systems, assuming the loss of one Class 1E Engineered Safety Features Actuation Cabinet (ESFAC) Division, which is the worst case single active failure, resulting in one Passive Containment Cooling System flow path being rendered inoperable.

APPLICABLE SAFETY ANALYSES (continued)

The limiting DBA for the maximum peak containment air temperature is a LOCA or SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 120°F.

The DBA temperature transients are used to establish the environmental qualification operating envelope for containment. The basis of the containment environmental qualification temperature envelope is to ensure the performance of safety related equipment inside containment (Ref. 2). The containment vessel design temperature is 300°F. The containment vessel temperature remains below 300°F for DBAs. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBAs.

The temperature limit is also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Passive Containment Cooling System (Ref. 1).

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 SLB or LOCA. The temperature limit is used in the DBA analyses 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(c)(2)(ii).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is computed to remain within acceptable limits. As a result, the ability of containment to perform its design function is ensured.

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 its 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 conservative 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 the containment average air temperature is within the LCO limit ensures that containment operation remains within the limits 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 associated containment atmosphere. The 24 hour Frequency of this Surveillance Requirement 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 main control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. Section 6.2, “Containment Systems.”

2. 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Passive Containment Cooling System (PCS) – Operating

BASES

BACKGROUND

The PCS provides containment cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA). The Passive Containment Cooling System is designed to meet the requirements of 10 CFR 50 Appendix A GDC 38 "Containment Heat Removal" and GDC 40 "Testing of Containment Heat Removal Systems" (Ref. 1).

The PCS consists of a 800,000 gal (nominal) cooling water tank, four headered tank discharge lines with flow restricting orifices, and two separate full capacity discharge flow paths to the containment vessel with 3 sets of isolation valves, each capable of meeting the design bases. Algae growth is not expected within the PCCWST; however, to assure water clarity is maintained, a prevailing concentration of hydrogen peroxide is maintained at 50 ppm. The recirculation pumps and heater provide freeze protection for the passive containment cooling water storage tank. However, OPERABILITY of the tank is assured by compliance with the temperature limits specified in SR 3.6.6.1 and not by the recirculation pumps and heater. In addition to the recirculation pumps and heater, the PCS water storage tank temperature can be maintained within limits by the ambient temperature, the large thermal inertia of the tank, or heat from other sources. The PCS valve room temperature must not be below freezing for an extended period to assure the water flow path to the containment shell is available. The isolation valves on each flow path are powered from a separate Division.

Upon actuation of the isolation valves, gravity flow of water from the cooling water tank (contained in the shield building structure above the containment) onto the upper portion of the containment shell reduces the containment pressure and temperature following a DBA. The flow of water to the containment shell surface is initially established to assure that the required short term containment cooling requirements following the postulated worst case LOCA are achieved. As the decay heat from the core becomes less with time, the water flow to the containment shell is reduced in three steps. The change in flow rate is attained without active components in the system and is dependent only on the decreasing water level in the elevated storage tank. In order to ensure the containment surface is adequately and effectively wetted, the water is introduced at the center of the containment dome and flows outward. Weirs are placed on the dome surface to distribute the water and ensure

BACKGROUND (continued)

effective wetting of the dome and vertical sides of the containment shell. The monitoring of the containment surface through the Reliability Assurance Program (RAP) and the Inservice Testing Program assures containment surface does not unacceptably degrade containment heat removal performance. During the initial test program, the containment coverage will be measured at the base of the upper annulus in addition to the coverage at the spring line for the full flow case and a lower flow case with PCS recirculation pumps delivering to the containment shell. These benchmark values at the base of the upper annulus will be used to develop acceptance criteria for technical specifications. Contamination can be removed by PCS actuation and by using coating vendor cleaning procedures.

The path for the natural circulation of air is from the air intakes in the shield building, down the outside of the baffle, up along the containment shell to the top, center exit in the shield building and is always open. The drains in the upper annulus region must be clear to prevent water from blocking the air flow path. Heat is removed from within the containment utilizing the steel containment shell as the heat transfer surface combining conductive heat transfer to the water film, convective heat transfer from the water film to the air, radiative heat transfer from the film to the air baffle, and mass transfer (evaporation) of the water film into the air. As the air heats up and water evaporates into the air, it becomes less dense than the cooler air in the air inlet annulus. This differential causes an increase in the natural circulation of the air upward along the containment surface, with heated air/water vapor exiting the top/center of the shield building. Additional system design details are provided in Reference 3.

The PCS is actuated either automatically, by a containment High-2 pressure signal, or manually. Automatic actuation opens the cooling water tank discharge valves, allowing gravity flow of the cooling water onto the containment shell. The manual containment cooling actuation consists of four momentary controls, if two associated controls are operated simultaneously actuation will occur in all divisions. The discharge continues for at least three days.

The PCS is designed to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a DBA.

The PCS is an ESF system and is designed to ensure that the heat removal capability required during the post accident period can be attained.

BASES

APPLICABLE SAFETY ANALYSES

The Passive Containment Cooling System limits the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF system, assuming the loss of one Class 1E Engineered Safety Features Actuation Cabinet (ESFAC) Division, which is the worst case single active failure and results in one PCS flow path being inoperable.

The analyses and evaluations assume a unit specific power level of 3400 MWt, one passive containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Passive Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment cooling system performance for post accident conditions is given in Reference 3. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post accident condition.

The modeled Passive Containment Cooling System actuation response time from the containment analysis is based upon a response time associated with exceeding the containment High-2 pressure setpoint to opening of isolation valves.

The Passive Containment Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, one passive containment cooling water flow path is required to maintain the containment peak pressure and temperature below the design limits (Ref. 3). To ensure that this requirement is met, two passive containment cooling water flow paths are provided.

BASES

LCO (continued)

Therefore, in the event of an accident, at least one flow path operates, assuming the worst case single active failure occurs. A third PCS flow path is provided for protection against multiple failure scenarios modeled in the PRA. To ensure that these requirements are met, three PCS water flow paths must be OPERABLE.

The PCS includes a cooling water tank, valves, piping, instruments and controls to ensure an OPERABLE flow path capable of delivering water from the cooling water tank upon an actuation signal. An OPERABLE flow path consists of a normally closed valve capable of automatically opening in series with a normally open valve. For the two flow paths containing air-operated valves, it is preferred because of PRA insights that these valves be normally closed.

The PCS cooling water storage tank ensures that an adequate supply of water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA). To be considered OPERABLE, the PCS cooling water storage tank must meet the water volume and temperature limits established in the SRs. To be considered OPERABLE, the air flow path from the shield building annulus inlet to the exit must be unobstructed, with unobstructed upper annulus safety-related drains providing a path for containment cooling water runoff to preclude blockage of the air flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the PCS.

During shutdown the PCS may be required to remove heat from containment. The requirements in MODES 5 and 6 are specified in LCO 3.6.7, Passive Containment Cooling System (PCS) – Shutdown.

ACTIONS

A.1

With one passive containment cooling water flow path inoperable, the affected flow path must be restored within 7 days. In this degraded condition, the remaining flow paths are capable of providing greater than 100% of the heat removal needs after an accident, even considering the worst single failure. The 7 day Completion Time was chosen in light of the remaining heat removal capability and the low probability of a DBA occurring during this period.

BASES

ACTIONS (continued)

B.1

With two passive containment cooling water flow paths inoperable, at least one affected flow path must be restored to OPERABLE status within 72 hours. In this degraded condition, the remaining flow path is capable of providing greater than 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen in light of the remaining heat removal capability and the low probability of DBA occurring during this period.

C.1

If the cooling water tank is inoperable, it must be restored to OPERABLE status within 8 hours. The tank may be declared inoperable due to low water level or temperature out of limits. The 8 hour Completion Time is reasonable based on the remaining heat removal capability of the system and the availability of cooling water from alternate sources.

D.1 and D.2

If any of the Required Actions and associated Completion Times are not met, or if the LCO is not met for reasons other than Condition A, B, or C, 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 84 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. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

This surveillance requires verification that the cooling water temperature is within the limits assumed in the accident analyses. The 7-day Frequency is adequate to identify a temperature change that would approach the temperature limits since the tank is large and temperature variations are slow.

The surveillance Frequency is increased to 24 hours in the event that the tank temperature approaches its limits; i.e., once temperature increases

BASES

SURVEILLANCE REQUIREMENTS (continued)

either to ≥ 100°F, or decreases to ≤ 50°F. Since the maximum tank temperature variation during the normal surveillance Frequency of 7 days is only about 1°F, the tank temperature cannot exceed its limits before the increased surveillance Frequency takes effect.

SR 3.6.6.2

Verification that the cooling water volume is above the required minimum ensures that a sufficient supply is available for containment cooling. Since the cooling water volume is normally stable and low level is indicated by a main control room alarm, a 7 day Frequency is appropriate and has been shown to be acceptable in similar applications.

SR 3.6.6.3

Verifying the correct alignment of power operated, and automatic valves, excluding check valves, in the Passive Containment Cooling System provides assurance that the proper flow paths exist for system 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 positions prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through control room instrumentation or a system walkdown, that valves capable of potentially 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 flow path. This Frequency has been shown to be acceptable through operating experience.

SR 3.6.6.4

This SR requires verification that each automatic isolation valve actuates to its correct position upon receipt of 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 these Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient 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 confirmed by operating experience) of the equipment. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.5

Periodic inspections of the PCS air flow path from the shield building annulus inlet to the exit ensure that it is unobstructed, the baffle plates are properly installed, and the upper annulus safety-related drains are unobstructed. Although there are no anticipated mechanisms which would cause air flow path or annulus drain obstruction and the effect of a missing air baffle section is small, it is considered prudent to verify this capability every 24 months. Additionally, the 24 month Frequency is based on the desire to perform this Surveillance under conditions that apply during a plant outage, on the need to have access to the locations, 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 in similar situations.

SR 3.6.6.6

This SR requires performance of a Passive Containment Cooling System test to verify system flow and water coverage capabilities. The system performance test demonstrates that the containment cooling capability assumed in accident analyses is maintained by verifying the flow rates via each standpipe and measurement of containment wetting coverage. The System Level Operability Testing Program provides specific test requirements and acceptance criteria. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The first refueling and 10 year Frequency is based on the ability of the more frequent surveillances to verify the OPERABILITY of the active components and features which could degrade with time.

REFERENCES

1. 10 CFR 50, Appendix A, “General Design Criteria for Nuclear Power Plants.”
2. 10 CFR 50, Appendix K, “ECCS Evaluation Models.”
3. Chapter 6.2, “Containment Systems.”

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Passive Containment Cooling System (PCS) – Shutdown

BASES

BACKGROUND

A description of the PCS is provided in the Bases for LCO 3.6.6, "Passive Containment Cooling System – Operating."

APPLICABLE SAFETY ANALYSES

The PCS limits the temperature and pressure that could be experienced during shutdown following a loss of decay heat removal.

For shutdown events, the Reactor Coolant System (RCS) sensible and decay heat removal requirements are reduced as compared to heat removal requirements for MODE 1, 2, 3, or 4 events. Therefore, the shutdown containment heat removal requirements are bounded by analyses of MODES 1, 2, 3, and 4 events. A discussion of MODES 1, 2, 3, and 4 DBAs is provided in the Bases for LCO 3.6.6, "Passive Containment Cooling System (PCS) – Operating."

The PCS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

For postulated shutdown events, one passive containment cooling water flow path is required to provide the required containment heat removal capability (Ref. 1). To ensure that this requirement is met, two passive containment cooling water flow paths are provided. Therefore, in the event of an accident, at least one flow path operates, assuming the worst case single active failure occurs. A third PCS flow path is provided for protection against multiple failure scenarios modeled in the PRA. To ensure that these requirements are met, three PCS water flow paths must be OPERABLE.

The PCS includes a cooling water tank, valves, piping, instruments and controls to ensure an OPERABLE flow path capable of delivering water from the cooling water tank upon an actuation signal.

The PCS cooling water storage tank ensures that an adequate supply of water is available to cool and depressurize the containment in the event of a loss of decay heat removal. To be considered OPERABLE, the PCS cooling water storage tank must meet the water volume and temperature limits established in the SRs. To be considered OPERABLE, the air flow path from the shield building annulus inlet to the exit must be unobstructed, with unobstructed upper annulus safety-related drains providing a path for containment cooling water runoff to preclude blockage of the air flow path.

BASES

APPLICABILITY

OPERABILITY of the PCS is required in either MODE 5 or 6 with the calculated reactor decay heat greater than 9 MWt for heat removal in the event of a loss of nonsafety decay heat removal capabilities.

With the decay heat less than 9 MWt, the decay heat can be easily removed from containment with air cooling alone. Confirmation of decay heat levels may be determined consistent with the assumptions and analysis basis of ANS 1979 plus 2 sigma or via an energy balance of the reactor coolant system.

The PCS requirements in MODES 1, 2, 3, and 4 are specified in LCO 3.6.6, "Passive Containment Cooling System (PCS) – Operating."

ACTIONS

A.1

With one passive containment cooling water flow path inoperable, the affected flow path must be restored within 7 days. In this degraded condition, the remaining flow paths are capable of providing greater than 100% of the heat removal needs after an accident, even considering the worst single failure. The 7 day Completion Time was chosen in light of the remaining heat removal capability and the low probability of a DBA occurring during this period.

B.1

With two passive containment cooling water flow paths inoperable, at least one affected flow path must be restored to OPERABLE status within 72 hours. In this degraded condition, the remaining flow path is capable of providing greater than 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen in light of the remaining heat removal capability and the low probability of an event occurring during this period.

C.1

If the cooling water tank is inoperable, it must be restored to OPERABLE status within 8 hours. The tank may be declared inoperable due to low water volume or temperature out of limits. The 8 hour Completion Time is reasonable based on the remaining heat removal capability of the system and the availability of cooling water from alternate sources.

D.1.1, D.1.2, and D.2

Action must be initiated if any of the Required Actions and associated Completion Times are not met, or if the LCO is not met for reasons other than Condition A, B, or C. If in MODE 5 with the RCS

BASES

ACTIONS (continued)

pressure boundary open and/or pressurizer level < 20%, action must be initiated, immediately, to increase the RCS level to a pressurizer level ≥ 20% and to close the RCS so that the PRHR HX operation is available. If in MODE 6, action must be initiated, immediately, to increase the refueling cavity water level ≥ 23 feet above the top of the reactor vessel flange. In both cases, the time to RCS boiling is maximized by maximizing the RCS inventory and maintaining RCS temperature as low as practical. Additionally, action to suspend positive reactivity additions is required to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

These Actions place the plant in a condition which maximize the time to actuation of the Passive Containment Cooling System, thus providing time for repairs or application of alternative cooling capabilities.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

The LCO 3.6.6 Surveillance Requirements (SR 3.6.6.1 through 3.6.6.6) are applicable. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.6.6 for a discussion of each SR.

REFERENCES

1. Section 6.2, “Containment Systems.”

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Containment Penetrations

BASES

BACKGROUND

Containment closure capability is required during shutdown operations when there is fuel inside containment. Containment closure is required to maintain within containment the cooling water inventory. Due to the large volume of the IRWST and the reduced sensible heat during shutdown, the loss of some of the water inventory can be accepted. Further, accident analyses have shown that containment closure capability is <u>not</u> required to meet offsite dose requirements. Therefore, containment does not need to be leak tight as required for MODES 1 through 4.

In MODES 5 and 6, 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 requirement for containment leak tightness, compliance with the Appendix J leakage criteria and tests are not required.

In MODES 5 and 6, there is no potential for steam release into the containment immediately following a accident. Pressurization of the containment could only occur after heatup of the IRWST due to PRHR HX operation (MODE 5 with RCS intact) or after heatup of the RCS with direct venting to the containment (MODE 5 with reduced RCS inventory or MODE 6 with the refueling cavity not fully flooded) or after heatup of the RCS and refueling cavity (MODE 6 with refueling cavity fully flooded). The time from loss of normal cooling until steam release to the containment for four representative sets of plant conditions is shown in Figure B 3.6.8-1 as a function of time after shutdown. Because local manual action may be required to achieve containment closure it is assumed that the containment hatches, air locks and penetrations must be closed prior to steaming into containment.

Figure B 3.6.8-1 provides allowable closure times for four representative sets of plant conditions. The time to steaming is dependent on various plant parameters (RCS temperature, IRWST temperature, etc.) and plant configuration (RCS Pressure Boundary Intact, RCS Open, etc.). Therefore, the actual representation of the time to steaming may be different than that provided in Figure B 3.6.8-1. In determining the minimum time to steaming, conservative assumptions regarding core decay heat, RCS configuration, and initial RCS inventory are used to minimize the calculated time to steaming. The curves are based on the core decay heat prior to refueling so that closure times are longer following the core reload.

BACKGROUND (continued)

As presented in Tables 54-1 and 54-4 of Reference 2, the most risk significant events during shutdown are events that lead to a loss of RNS cooling. Of these, the limiting events that lead to steaming to containment are the loss of shutdown cooling events, specifically:

- Loss of decay heat removal during drained conditions due to a failure of component cooling water or service water system;
- Loss-of-offsite power during drained conditions; and
- Loss of decay heat removal during drained conditions due to failure of the normal residual heat removal system.

These events are further discussed in Section 19.59.5 of Reference 1. Time to steaming is dependent on the postulated RCS configuration (intact versus open), and is based on the response of the plant considering features such as the operation of the 4th stage ADS valves if necessary, status of the upper internals, status of refueling cavity, etc. Conservative assumptions regarding these features are made in the determination of the minimum time to steaming. The time assumed in the PRA to close the penetrations before steaming to containment included 15 minutes for the diagnosis and decision-making time, in addition to the time required to physically complete the closure action.

The risk of overdraining the RCS has been significantly reduced in the AP1000 due to the automatic protection features associated with the hot leg level instruments which isolate letdown on low hot leg water level. Overdraining the RCS is no longer a significant contributor to core damage, as shown in Table 54-4 of Reference 2.

The assumptions used in determining the required closure time for the various containment openings should be conservative, and should be consistent with the plant operating procedures, staffing levels, and status of the containment openings. The evaluation should consider the ability to close the containment for the limiting loss of shutdown cooling event, and considering the possibility of a station blackout. In determining if containment can be closed within the time permitted to containment closure specified in Figure B 3.6.8 -1, the time to close containment penetrations must include both the diagnosis and decision-making time and the time required to physically complete the closure action.

Containment should be closed during the initial mid-loop period for a refueling since the time permitted to containment closure is shorter than the time to diagnose and make a decision that closure is needed

BASES

BACKGROUND (continued)

following an event. The need to close containment for the mid-loop period following a refueling must be evaluated since decay heat varies with the time after shutdown and the impact of the partial core replacement with new fuel. It is expected that containment will be closed for activities where drain-down is planned, such as the RCS drain-down from no-load pressurizer level for the initial mid-loop period during a refueling. Containment is not expected to be closed for minor, unplanned RCS volume transients, such as a short-term inventory where the pressurizer level may be reduced, but not emptied, and where recovery actions are within the time to containment closure.

The containment equipment hatches, which are part of the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. If closed, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that bolts required by this LCO be approximately equally spaced. Alternatively, if open, each equipment hatch can be installed using a dedicated set of hardware, tools and equipment. A self-contained power source is provided to drive each hoist while lowering the hatch into position. Large equipment and components may be moved through the hatches as long as they can be removed and the hatch closed prior to steaming into the containment.

The design of the equipment hatch is such that the four bolts would only be needed to support the hatch in place and provide adequate strength to support the hatch dead weight and associated loads. The hatch is installed on the inside containment and is held in place against a matching flange surface with mating bolt pattern by the bolts. Once the dead weight is supported, any pressure (greater than atmospheric) within containment will serve to exert closure force on the hatch toward the mating flange surface serving to reduce stresses on bolts. Therefore the determination of the number of bolts is limited to the quantity required to support the hatch itself and not related to any potential containment pressure.

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 operation 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 required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for

BASES

BACKGROUND (continued)

extended periods when frequent containment entry is necessary. Temporary equipment connections (e.g., power or communications cables) are permitted as long as they can be removed to allow containment closure prior to steaming into the containment.

Containment spare penetrations which also provide a part of the containment boundary provide for temporary support services (electrical, I&C, air, and water supplies) during MODES 5 and 6. Each penetration is flanged and normally closed. During periods of plant shutdown, temporary support systems may be routed through the penetrations; temporary equipment connections (e.g., power or communications cables) are permitted as long as they can be removed to allow containment closure prior to steaming into the containment. The spare penetrations must be closed or, if open, capable of closure prior to steaming to containment.

Containment penetrations, including purge system flow paths, that provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being 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 barrier for the containment penetrations. The equivalent isolation barrier must be capable of maintaining containment isolation at the containment design pressure of 59 psig (Ref. 1).

APPLICABLE SAFETY ANALYSES

For postulated shutdown events in MODES 5 and 6, RCS heat removal is provided by either passive residual heat removal (PRHR) or IRWST injection and containment sump recirculation. To support RCS heat removal, containment closure is required to limit the loss of the cooling water inventory from containment (Ref. 1).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO limits the loss of cooling water inventory in containment to assure continued coolant inventory by limiting the potential escape paths for water released within containment. Penetrations closed in accordance with these requirements are not required to be leak tight.

BASES

LCO (continued)

The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed or capable of being closed prior to steaming into the containment. The equipment hatches may be open; however, the hatches shall be clear of obstructions such that capability to close the hatch within the indicated time period is maintained. The hardware, tools, equipment and power sources necessary to install the hatches shall be available when the hatch is open. Both doors in each containment air lock may be open; however, the air locks shall be clear of obstructions such that the capability to close at least one door within the indicated time period is maintained. Alternatively, one door in an air lock may be closed. Containment spare penetrations may be open; however, the penetrations shall be capable of being closed within the indicated time period. Direct access penetrations shall be closed by at least one manual or automatic isolation valve, blind flange or equivalent, or capable of being closed by at least one valve actuated by a containment isolation signal. If direct access penetrations are open, OPERABILITY of the containment isolation instrumentation is required for the open penetrations by LCO 3.3.2, Function 3.a, Containment Isolation, Manual Initiation. An OPERABLE Containment Isolation Function includes LCO 3.3.2, Function 19.b, Containment Air Filtration System Isolation, Containment Isolation. Figure B 3.6.8-1 provides the acceptable required closure times for various representative MODES and conditions.

APPLICABILITY

The containment penetration requirements are applicable during conditions for which the primary safety related core cooling and boration capabilities are provided by IRWST or injection or PRHR – MODES 5 and 6. The capability to close containment is required to ensure that the cooling water inventory is not lost in the event of an accident.

In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1.

ACTIONS

A.1

If the containment equipment hatches, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the containment isolation function not capable of actuation when automatic isolation valves are open, the penetration(s) must be restored to the required status within 1 hour.

ACTIONS (continued)

B.1.1, B.1.2, and B.2

If Required Action A.1 is not completed within 1 hour or the LCO is not met for reasons other than Condition A, action must be taken to minimize the probability and consequences of an accident.

In MODE 5, action must be initiated, immediately, to be in MODE 5 with a pressurizer level ≥ 20% and to close the RCS so that the PRHR HX operation is available. In MODE 6, action must be initiated, immediately, to be in MODE 6 with the refueling cavity water level ≥ 23 feet above the top of the reactor vessel flange. The time to RCS steaming to containment is maximized by maximizing RCS inventory, and allowing PRHR HX operation. Additionally, action to suspend positive reactivity additions is required to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

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 purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal. Open containment spare penetrations shall be verified capable of being closed prior to steaming to containment by removal of obstructions and installation of the flange or by other closure means which will limit loss of the cooling water inventory from containment.

The Surveillance is performed every 7 days. The Surveillance interval is selected to ensure that the required penetration status is maintained during shutdown inspections, testing, and maintenance.

SR 3.6.8.2

Each of the two equipment hatches is provided with a set of hardware, tools, equipment, and self-contained power source for moving the hatch from its storage location and installing it in the opening. The required set of hardware and tools shall be visually inspected to ensure that they can perform the required functions. The equipment and power source shall

BASES

SURVEILLANCE REQUIREMENTS (continued)

be inspected and/or operated as necessary to verify that the hatch can be installed. The power source shall be verified as containing sufficient energy to install the hatch from the storage location.

The 7 day Frequency is adequate considering that the hardware, tools, equipment, and power sources are dedicated to the associated equipment hatch and not used for any other functions.

The SR is modified by a Note which only requires that the surveillance be met for an open equipment hatch. If the equipment hatch is installed in position, then the availability of the means to install the hatch is not required.

SR 3.6.8.3

This Surveillance demonstrates that at least one valve in each open penetration actuates to its isolation position on manual initiation or on an actual or simulated containment isolation signal. The 24 month Frequency maintains consistency with other similar valve testing requirements. The OPERABILITY requirements for the Containment Isolation function are specified in LCO 3.3.2.

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.

REFERENCES

1. DCD Chapter 19.
2. AP1000 Probabilistic Risk Assessment.

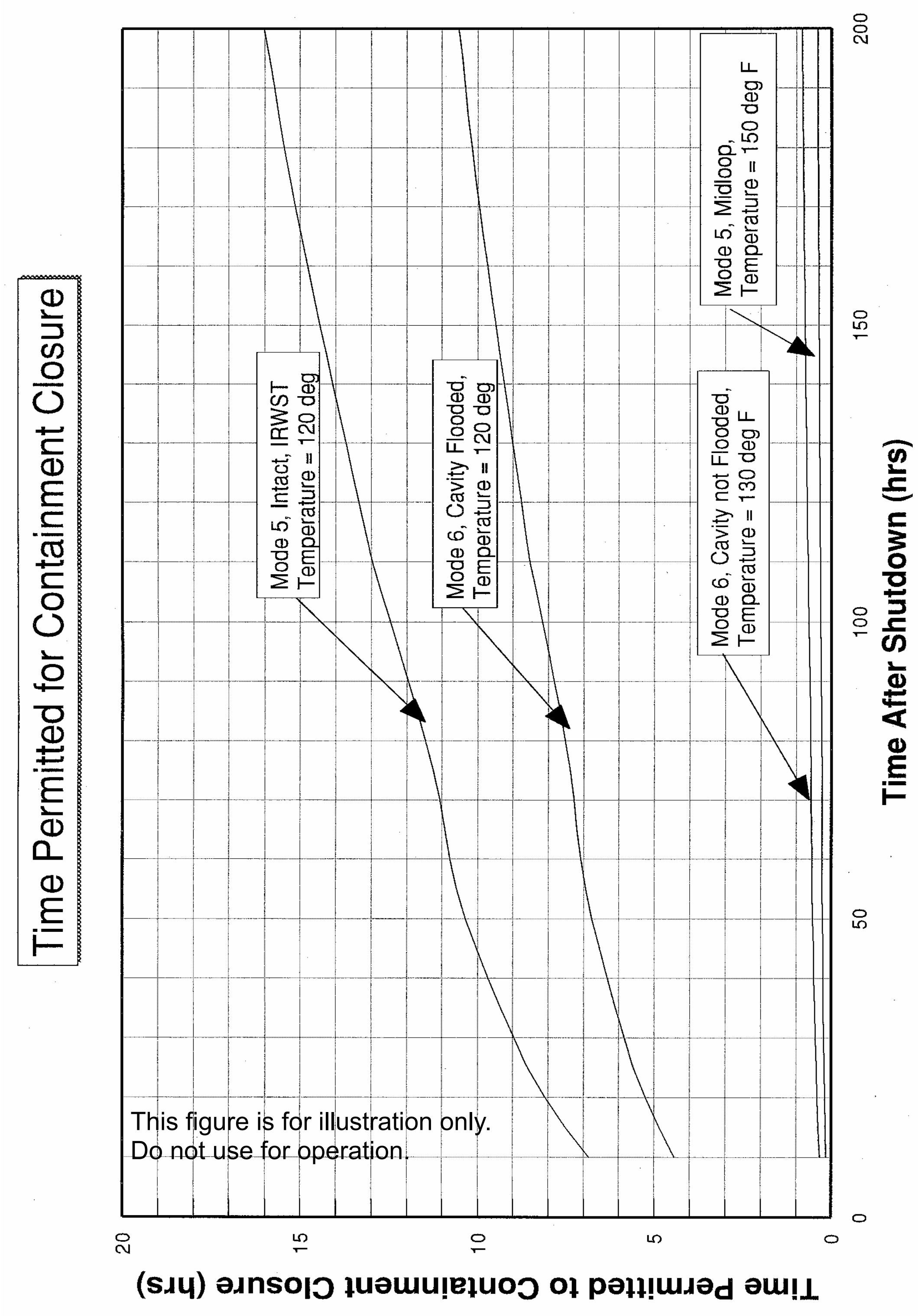

Figure B 3.6.8-1 (Page 1 of 1)
Time Prior to Coolant Inventory Boiling

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 pH Adjustment

BASES

BACKGROUND

The Passive Core Cooling System (PXS) includes two pH adjustment baskets which 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. Chemical addition is necessary to counter the affects of the boric acid contained in the safety injection supplies and acids produced in the post-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 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 at an elevation that is below the minimum floodup level. The baskets are placed at least a foot above the floor to reduce the chance that water spills will dissolve the TSP. Natural recirculation of water inside the containment, following a LOCA, is driven by the core decay heat and provides mixing to achieve a uniform 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. (Refs. 1 and 2)

APPLICABLE SAFETY ANALYSES

In the event of a Design Basis Accident (DBA), iodine may be released from the fuel to containment. To limit this iodine release from containment, the pH of the water in the containment sump is adjusted by the addition of TSP. Adjusting the sump water 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.

pH adjustment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The requirement to maintain the pH adjustment baskets with ≥ 560 ft^3 of TSP assures that for DBA releases of iodine into containment, the pH of the containment sump will be adjusted to enhance the retention of the iodine.

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. 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

A.1

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.

B.1 and B.2

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 84 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 3.6.9.1

The minimum amount of TSP is 560 ft^3. A volume is specified since it is not feasible to weigh the TSP contained in the pH adjustment baskets. 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 (908,000 gallons) containing the maximum amount of boron (2990 ppm) as well as other sources of acid. The minimum required mass of TSP is 26,460 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 about 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.

SR 3.6.9.2

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of 2.39 grams of TSP from one of the baskets in containment is submerged in ≥ 1 liter of water at a boron concentration of 2990 ppm and at the standard temperature of 25 ± 5°C. Without agitation, the solution pH should be raised to ≥ 7.0 within 4 hours.

The minimum required amount of TSP is sufficient to buffer the maximum amount of boron 2990 ppm, the maximum amount of other acids, and the maximum amount of water 908,000 gallons that can exist in the containment following an accident and achieve a minimum pH of 7.0.

Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved TSP to naturally diffuse through the sample solution. In the post LOCA sump area, rapid mixing would occur due to liquid flow, significantly decreasing the actual amount of time

BASES

SURVEILLANCE REQUIREMENTS (continued)

before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH $\geq$ 7.0 by the onset of recirculation after a LOCA.

REFERENCES

1. Section 6.3.2.1.4, “Containment pH Control.”
2. Section 6.3.2.2.4, “pH Adjustment Baskets.”
3. Section 15.6.5.3.1, “Identification of Cause and Accident Description.”

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing 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 and Circulating Water System, is not available.

Six MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in Reference 1. The MSSVs 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 design includes staggered setpoints, as shown in Table 3.7.1-2 of the specification, so that only the needed valves actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open the valves following a turbine-reactor trip.

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 for any anticipated operating occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, are those characterized as decreased heat removal events, which are presented in Section 15.2 (Ref. 3). Of these, the full power turbine trip without turbine bypass is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for turbine trip without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer spray. This analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure.

APPLICABLE SAFETY ANALYSES (continued)

All cases analyzed demonstrate that the 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. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The DCD Section 15.4.2 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The DCD safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value.

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.

APPLICABLE SAFETY ANALYSES (continued)

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The accident analysis requires six MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% of RTP. A MSSV will be considered inoperable if it fails to open in the event of a pressure excursion in excess of the setpoint. The LCO requires that six MSSVs be OPERABLE in compliance with Reference 2. Operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1-1 of the specification and Required Action A.1.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings specified in Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODE 1, 2, 3, or 4 (without the normal residual heat removal system in service), six MSSVs per steam generator are required to be OPERABLE.

In MODES 4 (with the normal residual heat removal system in service) 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 MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

With one or more MSSVs inoperable, reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all six MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 6, with an appropriate allowance for calorimetric power uncertainty.

To determine the maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs, the governing heat transfer relationship is the equation $q = \dot{m}\Delta h$, where q is the heat input from the primary side, $\dot{m}$ is the mass flow rate of the steam, and Δh is the increase in enthalpy that occurs in converting the secondary side water to steam. If it is conservatively assumed that the secondary side water is all saturated liquid (i.e., no subcooled feedwater), then the Δh is the heat of vaporization (h_{fg}) at the steam relief pressure. The following equation is used to determine the maximum allowable power level for continued operation with inoperable MSSVs.

$$\text{Maximum NSSS Power} \leq (100/Q)\ (W_s\ h_{fg}\ N)\ /\ K$$

where:

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), MWt

K = Conversion factor, 947.82 (Btu/sec)/MWt

w_s = Minimum total steam flow rate capability of the OPERABLE MSSVs on any one steam generator at the highest OPERABLE MSSV opening pressure, including tolerance and accumulation as appropriate, lbm/sec

ACTIONS (continued)

h_{fg} = Heat of vaporization at the highest MSSV opening pressure, including tolerance and accumulation as appropriate, Btu/lbm

N = Number of steam generators in the plant

To determine the Table 3.7.1-1 Maximum Allowable Power, the Maximum NSSS Power calculated using the equation above is reduced by 9% RTP to account for Nuclear Instrument System trip channel uncertainties.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4, with RCS cooling provided by the RNS, 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 plant 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 safety and relief valve test are required to be performed in accordance with ASME OM Code (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Set pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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. Table 3.7.1-2 allows a ±3% setpoint tolerance

for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, 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.

REFERENCES

1. Chapter 10, “Steam and Power Conversion Systems Description.”
2. ASME Boiler and Pressure Vessel Code, Section III, Article NC-7000, “Overpressure Protection,” Class 2 Components.
3. Section 15.2, “Decreased Heat Removal by Secondary System.”
4. ASME Boiler and Pressure Vessel Code, Section XI, Article IV-3500, “Inservice Test: Category C Valves.”
5. ASME OM Code-1995 and Addenda through the 1996 Addenda, “Code for Operation and Maintenance of Nuclear Power Plants.”
6. NRC Information Notice 94-60, “Potential Overpressurization of the Main Steam System,” August 22, 1994.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

Each main steamline has one safety related MSIV to isolate steam flow from the secondary side of the steam generators following a high energy line break. MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside containment. The MSIVs are downstream from the main steam safety valves (MSSVs). Downstream from the MSIVs, main steam enters the high pressure turbine through four stop valves and four governing control valves. Closing the MSIVs isolates each steam generator from the other and isolates the turbine bypass system, and other steam supplies from the steam generator.

The MSIVs, turbine stop and control valves, turbine bypass valves, and moisture separator reheater 2nd stage steam isolation valves close on a main steam isolation signal generated by either low steam line pressure, high containment pressure, Low T_{cold}, or high negative steam pressure rate. The MSIVs fail closed on loss of control air or actuation signal from either of two 1E power divisions.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the Section 10.3 (Ref. 1). Descriptions for the turbine bypass valves, and moisture separator reheater 2nd stage steam isolation valves are found in the Section 10.4 (Ref. 6).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the Section 6.2 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the Section 15.1 (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).

Design basis events of concern for containment analysis are SLB inside containment with the failure of the associated MSIV to close, or a main feedline break with the associated failure of a feedline isolation or control

BASES

APPLICABLE SAFETY ANALYSES (continued)

valve to close. At lower powers, 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 MSIV, 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 core is ultimately shut down by the boric acid injection delivered by the Core Makeup Tanks (CMTs).

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment 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 large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes consideration of scenarios with offsite power available, and with a loss of offsite power. With offsite power available, the reactor coolant pumps continue to circulate coolant for a longer period through the steam generators, maximizing the Reactor Coolant System cooldown. The reactor protection system includes a safety related signal that initiates the coastdown of the reactor coolant pumps early in the large SLB transient. Therefore, there is very little difference in the predicted departure from nucleate boiling ratio between cases with and without offsite power. Significant single failures considered include failure of an MSIV to close.

The non-safety related turbine stop or control valves, in combination with the turbine bypass, and moisture separator reheater 2^{nd} stage steam isolation valves, are assumed as a backup to isolate the steam flow path given a single failure of an MSIV. The safety analyses do not differentiate between the availability of the turbine stop valve or its series control valve. Either the turbine stop valves or its associated turbine control valve are required by this LCO to be OPERABLE. These valves, along with the turbine bypass, and moisture separator reheater 2^{nd} stage steam isolation valves are considered as alternate downstream valves.

The MSIVs serve a safety related function and remain open during power operation. These valves operate under the following situations:

a. High energy line break inside containment. 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 both steam generators until the unaffected loop MSIV closes. After

APPLICABLE SAFETY ANALYSES (continued)

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 MSIV in the unaffected loop. Closure of the MSIV isolates the break from the unaffected steam generator.

b. A break outside of containment, and upstream or downstream 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 or alternate downstream valves isolates the break, and limits the blowdown to a single steam generator.

c. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator to minimize radiological releases.

d. The MSIVs are also utilized during other events such as a feedwater line break; however, these events are less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs and the alternate downstream valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Following an SLB and main steam isolation signal, the analyses assume continued steam loss through the steamline condensate drain lines, turbine gland seal system, and the main steam to auxiliary steam header which supplies the auxiliary steam line to the deaerator. Since these valves are not assumed for steam isolation, they do not satisfy the 10 CFR 50.36(c)(2)(ii) criteria.

LCO

This LCO requires that one MSIV in each of the two steam lines be OPERABLE. The MSIVs are considered OPERABLE when their isolation times are within limits, and they close on an isolation actuation signal.

This LCO requires that four turbine stop valves or their associated turbine control valve, six turbine bypass valves, and two moisture separator reheater 2nd stage steam isolation valves be OPERABLE. A valve is considered OPERABLE when its isolation time is within the safety analysis isolation time limit of 5 seconds and it closes on an MSIV actuation signal. The

BASES

LCO (continued)

turbine bypass valves are alternatively considered OPERABLE when closed and administratively maintained closed with automatic actuation blocked as appropriate.

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 limits or the NRC staff approved licensing basis.

This LCO provides assurance that the design and performance of the alternate downstream valves are compatible with the accident conditions for which they are called upon to function (Ref. 5).

APPLICABILITY

The MSIVs, turbine stop or associated turbine control valves, turbine bypass valves, and moisture separator reheater 2nd stage steam isolation valves must be OPERABLE in MODE 1 and MODES 2, 3, and 4, except when steam flow is isolated when there is significant mass and energy in the RCS and steam generators. Therefore, these valves must be OPERABLE or closed. When these valves are closed, they are already performing their required function.

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 and alternate downstream valves are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the valves can be made with the plant hot. The 8 hour Completion Time is reasonable considering the low probability of an accident occurring during this time period that would require a closure of these valves. With a single MSIV inoperable, the safety function, isolation of the steam flow path, is provided by the OPERABLE alternate downstream valves, but cannot accommodate a single failure. The assumptions and criteria of the accident analyses are preserved by the ability to automatically isolate the steam flow path.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a

BASES

ACTIONS (continued)

closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides a positive means for containment isolation.

B.1

With any number of the turbine stop valves and the associated turbine control valve, turbine bypass, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 1, action must be taken to restore OPERABLE status within 72 hours. Some repairs to the valves can be made with the plant hot. The 72 hour Completion Time is reasonable considering the low probability of an accident occurring during this time period that would require a closure of these valves. With the backup isolation valves inoperable, the safety function, isolation of the steam flow path, is provided by the remaining OPERABLE valves, but cannot accommodate a single failure. The assumptions and criteria of the accident analyses are preserved by the ability to automatically isolate the steam flow path.

C.1

With two MSIVs inoperable in MODE 1 or one MSIV and an alternate downstream valve inoperable or if the valves cannot be restored to OPERABLE status in accordance with Required Action A.1 or B.1, 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 Time is reasonable, based on operating experience, to reach MODE 2 in an orderly manner and without challenging unit systems.

D.1 and D.2

Condition D is modified by a Note indicating that a separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2, 3, and 4, 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 consistent with that allowed in Condition A, and conservative considering the reduced energy in the steam generators in MODES 2, 3, and 4.

BASES

ACTIONS (continued)

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time but were closed, these inoperable valves must be verified to be continually closed on a periodic basis. This is necessary to ensure that the assumptions in the safety analyses remain valid. The 7 day Completion Time is based on engineering judgment, and is considered reasonable in view of MSIV status indications available in the control room and other administrative controls which ensure that these valves will continue to be closed.

E.1 and E.2

If the MSIVs cannot be restored to OPERABLE status or closed within the associated Completion Times of Condition D, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 with normal residual heat removal system in service within 24 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5.0 seconds, on an actual or simulated actuation signal. The MSIV isolation time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME OM Code (Ref. 7), 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.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

This SR verifies that the turbine stop, turbine control, turbine bypass, and moisture separator reheater 2nd stage steam isolation valves' closure time is ≤ 5.0 seconds, on an actual or simulated actuation signal. These alternate downstream isolation valves must meet the MSIV isolation time assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The alternate downstream valves should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the alternate downstream valves are not tested at power, they are exempt from the ASME OM Code (Ref. 7), 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.

REFERENCES

1. Section 10.3, "Main Steam System."
2. Section 6.2.1, "Containment Functional Design."
3. Section 15.1, "Increase in Heat Removal by Secondary System."
4. Section 10.2, "Turbine Generator."
5. NUREG-138, Issue 1, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director NRR to NRR Staff."
6. Section 10.4, "Other Features of Steam and Power Conversion Systems."
7. ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation and Control Valves (MFIVs and MFCVs)

BASES

BACKGROUND

The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break. The safety related function of the MFCVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following a high energy line break. Closure of the MFIVs or MFCVs terminates flow to the steam generators, terminating the event for feedwater line breaks occurring upstream of the MFIVs or MFCVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs or MFCVs, effectively terminates the addition of main feedwater to an affected steam generator, limiting the mass and energy release for steam or feedwater line breaks inside containment, and reducing the cooldown effects for steam line breaks (SLBs).

The MFIVs or MFCVs isolate the nonsafety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of startup feedwater (SFW) to the intact loops of the steam generator.

One MFIV and one MFCV are located on each MFW line, outside but close to containment. The MFIVs and MFCVs are located in the MFW line and are independent of the delivery of the MFW or SFW via the SFW line which is separately connected and isolated from the steam generator. This configuration permits MFW or SFW to be supplied to the steam generators following MFIV or MFCV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases following either an SLB or FWLB.

The MFIVs and MFCVs close on receipt of engineered safeguards feedwater isolation signal generated from any of the following conditions:

- Automatic or manual safeguards actuation “S” signal
- High steam generator level
- Low-2 T_{avg} signal coincident with reactor trip (P-4)
- Manual actuation

BASES

BACKGROUND (continued)

Additionally, the MFIVs close automatically on a Low-1 T_{avg} coincident with reactor trip (P-4). Each valve may be actuated manually. In addition to the MFIVs and the MFCVs, a check valve is available outside containment to isolate the feedwater line penetrating containment. In the event of feedwater line depressurization due to pump trip on line break, the check valve provides rapid backup isolation of the steam generators limiting the inventory loss. A description of the MFIVs and MFCVs is found in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs and MFCVs is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large Feedwater Line Break (FWLB). Closure of the MFIVs (or MFCVs) may also be relied on to mitigate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level – High 2 signal.

Failure of an MFIV (or MFCV), to close following an SLB or FWLB, can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs and MFCVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures that the MFIVs and the MFCVs will isolate the main feedwater system.

This LCO requires that the one isolation valve and one control valve on each feedwater line be OPERABLE. These valves are considered OPERABLE when their 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 following an SLB or FWLB inside containment. A main feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, and therefore failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

The MFIVs and MFCVs must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and the steam generators. This ensures that, in the event of a high energy line

APPLICABILITY (continued)

break, a single failure cannot result in the blowdown of more than one steam generator. In MODE 1, 2, 3, or 4, these valves are required to be OPERABLE to limit the amount of available fluid that could be added to the containment in the case of a secondary system pipe break inside containment. When the valves are closed and deactivated or isolated by a closed manual valve, they are already performing their safety function.

In MODES 5 and 6 steam generator energy is low. Therefore, the MFIVs and the MFCVs 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 valve.

A.1, A.2, B.1, and B.2

With one or two MFIVs, or one or two MFCVs inoperable, close or isolate inoperable affected flow path in 72 hours. When these flow paths are isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, and the low probability of an event that would require isolation of the main feedwater flow paths occurring during this period.

For inoperable MFIVs and MFCVs valves that cannot be restored to OPERABLE status within the specified Completion Time but are closed or isolated, the flow paths must be verified on a periodic basis to be closed or isolated. This is necessary to ensure that the assumptions in the safety analyses 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 or isolated.

C.1

With two inoperable valves in the same flow path there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, one valve in the affected flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the situation in which at least one valve in the affected flow path is performing the required safety function. The 8 hour Completion Time is a reasonable amount of time to complete the actions required to close the MFIV, or

BASES

ACTIONS (continued)

MFCV, which includes performing a controlled plant shutdown. The Completion Time is reasonable based on operating experience to reach MODE 2 with the MFIV or MFCV closed, from full-power conditions in an orderly manner and without challenging plant systems.

D.1, D.2, and D.3

If the MFIVs and MFCVs cannot be restored to OPERABLE status, or closed, or isolated 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 plant must be placed in at least MODE 3 within 6 hours, in MODE 4 with the normal residual heat removal system in service within 24 hours, and the affected flow path isolated within 36 hours or 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV and MFCV is ≤ 5.0 seconds, on an actual or simulated actuation signal. The MFIV and MFCV isolation times are assumed in the accident and containment analyses. This Surveillance 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 when the unit is generating power. This is consistent with the ASME OM Code (Ref. 2), quarterly stroke requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

The 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.

REFERENCES

1. Section 10.4.7, “Condensate and Feedwater System.”

2. ASME OM Code, ”Code for Operation and Maintenance of Nuclear Power Plants.”

B 3.7 PLANT SYSTEMS

B 3.7.4 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube LEAKAGE from the Reactor Coolant System (RCS). Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant. While fission products present in the primary coolant, as well as activated corrosion products, enter the secondary coolant system due to the primary to secondary LEAKAGE, only the iodines are of a significant concern relative to airborne release of activity in the event of an accident or abnormal occurrence (radioactive noble gases that enter the secondary side are not retained in the coolant but are released to the environment via the condenser air removal system throughout normal operation).

The limit on secondary coolant radioactive iodines minimizes releases to the environment due to anticipated operational occurrences or postulated accidents.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (SLB) as discussed in Chapter 15 (Ref. 1) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.1 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 a postulated SLB are within the acceptance criteria in SRP Section 15.0.1, and within the exposure guideline values of 10 CFR Part 50.34.

Secondary specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit of the secondary coolant is required to be ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131 to maintain the validity of the analyses reported in Chapter 15 (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 DBA.

BASES

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.

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 3.7.4.1

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. Chapter 15, “Accident Analyses.”

B 3.7 PLANT SYSTEMS

B 3.7.5 Spent Fuel Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel 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 at their maximum capacity. The water also provides shielding during the movement of spent fuel, and a large capacity heat sink in the event the spent fuel pool cooling system is inoperable.

A general description of the spent fuel pool design is given in Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling System is given in Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in Section 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The design basis radiological consequences resulting from a postulated fuel handling accident are within the dose values provided in Section 15.7.4 (Ref. 3).

According to Reference 3 there is 23 ft of water between the damaged fuel bundle and the fuel pool surface during a fuel handling accident. 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. This slight reduction in water depth does not adversely affect the margin of conservatism associated with the assumed pool scrubbing factor of 500 for elemental iodine.

In addition to mitigation of the effects of a fuel handling accident, the required minimum water level in the spent fuel pool provides a large capacity heat sink for spent fuel pool cooling in the event the spent fuel pool cooling system is inoperable.

The Spent Fuel Pool Water Level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The spent fuel 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

BASES

LCO (continued)

analysis (Ref. 3) and loss of spent fuel pool cooling. As such, it is the minimum required for fuel storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies at all times since the loss of spent fuel pool cooling is not MODE dependent.

ACTIONS

LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since spent fuel pool cooling requirements apply at all times, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. Spent fuel pool cooling requirements are 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.

LCO 3.0.8 is applicable while in MODE 5 or 6. Since spent fuel pool cooling requirements apply at all times, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. Spent fuel pool cooling requirements are independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

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 pool water level is lower than the required level, the movement of irradiated fuel assemblies shall be suspended. 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 4, 5, or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2 and 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

A.2

If the water level in the spent fuel pool is < 23 ft, the heat capacity of the spent fuel pool will be less than that assumed in the event of a loss of spent fuel pool cooling. In this case, action must be initiated within 1 hour

BASES

ACTIONS (continued)

to restore the water level in the spent fuel pool to ≥ 23 ft above the top of the irradiated fuel assemblies. Initiation of this action requires that the action be continued until a water level of ≥ 23 ft is attained.

The Completion Time of 1 hour assures prompt action to compensate for a degraded condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR verifies sufficient spent fuel pool water is available in the event of a fuel handling accident or loss of spent fuel pool cooling. The water level in the spent fuel 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 pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.4.1.

REFERENCES

1. Section 9.1.2, “Spent Fuel Storage.”
2. Section 9.1.3, “Spent Fuel Pool Cooling System.”
3. Section 15.7.4, “Fuel Handling Accident.”
4. Regulatory Guide 1.183 Rev. 0, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Control Room Emergency Habitability System (VES)

BASES

BACKGROUND

The Main Control Room Habitability System (VES) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. The system is designed to operate following a Design Basis Accident (DBA) which requires protection from the release of radioactivity. In these events, the Nuclear Island Non-Radioactive Ventilation System (VBS) would continue to function if AC power is available. If AC power is lost or a High-2 main control room (MCR) radiation signal is received, the VES is actuated. The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air (Ref. 4) for the MCR occupants; 2) to provide forced ventilation to maintain the MCR at a 1/8 inch water gauge positive pressure with respect to the surrounding areas; and 3) to limit the temperature increase of the MCR equipment and facilities that must remain functional during an accident, via the heat absorption of passive heat sinks.

The VES consists of compressed air storage tanks, two air delivery flow paths, associated valves, piping, and instrumentation. The tanks contain enough breathable air to supply the required air flow to the MCR for at least 72 hours. The VES system is designed to maintain CO_2 concentration less than 0.5% for up to 11 MCR occupants.

Sufficient thermal mass exists in the surrounding concrete structure (including walls, ceiling and floors) to absorb the heat generated inside the MCR, which is initially at or below 75°F. Heat sources inside the MCR include operator workstations, emergency lighting and occupants. Sufficient insulation is provided surrounding the MCR pressure boundary to preserve the minimum required thermal capacity of the heat sink. The insulation also limits the heat gain from the adjoining areas following the loss of VBS cooling.

In the unlikely event that power to the VBS is unavailable for more than 72 hours, MCR envelope habitability is maintained by operating one of the two MCR ancillary fans to supply outside air to the MCR envelope.

The compressed air storage tanks are initially pressurized to 3400 psig. During operation of the VES, a self contained pressure regulating valve maintains a constant downstream pressure regardless of the upstream pressure. An orifice downstream of the regulating valve is used to control the air flow rate into the MCR. The MCR is maintained at a 1/8 inch water gauge positive pressure to minimize the infiltration of airborne contaminants from the surrounding areas.

BASES

APPLICABLE SAFETY ANALYSES

The compressed air storage tanks are sized such that the set of tanks has a combined capacity that provides at least 72 hours of VES operation.

Operation of the VES is automatically initiated by the following safety related signals: 1) high-2 particulate or iodine radioactivity, or 2) low pressurizer pressure.

In the event of a loss of all AC power, the VES functions to provide ventilation, pressurization, and cooling of the MCR pressure boundary.

In the event of a high level of gaseous radioactivity outside of the MCR, the VBS continues to operate to provide pressurization and filtration functions. The MCR air supply downstream of the filtration units is monitored by a safety related radiation detector. Upon high-2 particulate or iodine radioactivity setpoint, or low pressurizer pressure, a safety related signal is generated to isolate the MCR from the VBS and to initiate air flow from the VES storage tanks. Isolation of the VBS consists of closing safety related valves in the supply and exhaust ducts that penetrate the MCR pressure boundary. VES air flow is initiated by a safety related signal which opens the isolation valves in the VES supply lines.

The VES functions to mitigate a DBA or transient that either assumes the failure of or challenges the integrity of the fission product barrier.

The VES satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The VES limits the MCR temperature rise and maintains the MCR at a positive pressure relative to the surrounding environment.

Two air delivery flow paths are required to be OPERABLE to ensure that at least one is available, assuming a single failure.

The VES is considered OPERABLE when the individual components necessary to deliver a supply of breathable air to the MCR are OPERABLE. This includes components listed in SR 3.7.6.2 through 3.7.6.8. In addition, the MCR pressure boundary must be maintained, including the integrity of the walls, floors, ceilings, electrical and mechanical penetrations, and access doors.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

LCO (continued)

The LCO is modified by a Note allowing the control room boundary 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 control room isolation is indicated.

APPLICABILITY

The VES is required to be OPERABLE in MODES 1, 2, 3, and 4 and during movement of irradiated fuel because of the potential for a fission product release following a DBA.

The VES is not required to be OPERABLE in MODES 5 and 6 when irradiated fuel is not being moved because accidents resulting in fission product release are not postulated.

ACTIONS

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

When a VES valve or damper is inoperable, action is required to restore the component to OPERABLE status. A Completion Time of 7 days is permitted to restore the valve or damper to OPERABLE status before action must be taken to reduce power. The Completion Time of 7 days is based on engineering judgment, considering the low probability of an accident that would result in a significant radiation release from the fuel, the low probability of not containing the radiation, and that the remaining components can provide the required capability.

B.1

When the main control room air temperature is outside the acceptable range during VBS operation, action is required to restore it to an acceptable range. A Completion Time of 24 hours is permitted based

ACTIONS (continued)

upon the availability of temperature indication in the MCR. It is judged to be a sufficient amount of time allotted to correct the deficiency in the nonsafety ventilation system before shutting down.

C.1

If the MCR pressure boundary is damaged or otherwise degraded, action is required to restore the integrity of the pressure boundary and restore it to OPERABLE status within 24 hours. A Completion Time of 24 hours is permitted based upon operating experience. It is judged to be a sufficient amount of time allotted to correct the deficiency in the pressure boundary.

D.1 and D.2

In MODE 1, 2, 3, or 4 if Conditions A, B, or C cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE that minimizes accident risk. This is done by entering MODE 3 within 6 hours and MODE 5 within 36 hours.

E.1

During movement of irradiated fuel assemblies, if the Required Action A.1, B.1, or C.1 cannot be completed within the required Completion Time, the movement of fuel must be suspended. Performance of Required Action E.1 shall not preclude completion of actions to establish a safe condition.

F.1, F.2, and F.3

If the VES is inoperable in MODE 1, 2, 3, or 4, the VES may not be capable of performing the intended function, and the plant must be brought to MODE 4, where the probability and consequences of an event are minimized, and the VES must be restored to OPERABLE status within 36 hours. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 within 12 hours.

G.1

During movement of irradiated fuel assemblies with the VES inoperable, the Required Action is to immediately suspend activities that present a potential for releasing radioactivity that might enter the MCR. This places the plant in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

The MCR air temperature is checked at a frequency of 24 hours to verify that the VBS is performing as required to maintain the initial condition temperature assumed in the safety analysis, and to ensure that the MCR temperature will not exceed the required conditions after loss of VBS cooling. The surveillance limit of 75°F is the initial heat sink temperature assumed in the VES thermal analysis. The 24 hour Frequency is acceptable based on the availability of temperature indication in the MCR.

SR 3.7.6.2

Verification every 24 hours that compressed air storage tanks are pressurized to ≥ 3400 psig is sufficient to ensure that there will be an adequate supply of breathable air to maintain MCR habitability for a period of 72 hours. The Frequency of 24 hours is based on the availability of pressure indication in the MCR.

SR 3.7.6.3

VES air delivery isolation valves are required to be verified as OPERABLE. The Frequency required is in accordance with the Inservice Testing Program.

SR 3.7.6.4

VES air header isolation valves are required to be verified open at 31 day intervals. This SR is designed to ensure that the pathways for supplying breathable air to the MCR are available should loss of VBS occur. These valves should be closed only during required testing or maintenance of downstream components, or to preclude complete depressurization of the system should the VES isolation valves in the air delivery line open inadvertently or begin to leak.

SR 3.7.6.5

Verification that the air quality of the air storage tanks meets the requirements of Appendix C, Table C-1 of ASHRAE Standard 62 is required every 92 days. If air has not been added to the air storage tanks since the previous verification, verification may be accomplished by confirmation of the acceptability of the previous surveillance results along with examination of the documented record of air makeup. The purpose of ASHRAE Standard 62 states: "This standard specifies minimum ventilation rates and indoor air quality that will be acceptable to human occupants and are intended to minimize the potential for adverse health

SURVEILLANCE REQUIREMENTS (continued)

effects." Verification of the initial air quality (in combination with the other surveillances) ensures that breathable air is available for 11 MCR occupants for at least 72 hours.

SR 3.7.6.6

Verification that all VBS isolation valves are OPERABLE and will actuate upon demand is required every 24 months to ensure that the MCR can be isolated upon loss of VBS operation.

SR 3.7.6.7

Verification that each VES pressure relief isolation valve within the MCR pressure boundary is OPERABLE is required in accordance with the Inservice Testing Program. The SR is used in combination with SR 3.7.6.7 to ensure that adequate vent area is available to mitigate MCR overpressurization.

SR 3.7.6.8

Verification that the VES pressure relief damper is OPERABLE is required at 24 month intervals. The SR is used in combination with SR 3.7.6.6 to ensure that adequate vent area is available to mitigate MCR overpressurization.

SR 3.7.6.9

Verification of the OPERABILITY of the self-contained pressure regulating valve in each VES air delivery flow path is required in accordance with the Inservice Testing Program. This is done to ensure that a sufficient supply of air is provided as required, and that uncontrolled air flow into the MCR will not occur.

SR 3.7.6.10

Per Reference 1, a functional test is required to establish that one VES air delivery flow path, using the safety related compressed air storage tanks, pressurizes the MCR envelope to at least a positive 1/8 inch water gauge pressure relative to the surrounding spaces at the required air addition flow rate of 65 ± 5 scfm (Ref. 3). The test need not last 72 hours, only long enough to demonstrate the ability to achieve the required differential pressure. The MCR envelope leakage rate must be within the design capacity of the VES to pressurize the MCR for 72 hours. One air

BASES

SURVEILLANCE REQUIREMENTS (continued)

delivery flow path is tested on an alternating basis. The system performance test demonstrates that the MCR pressurization assumed in dose analysis is maintained.

REFERENCES

1. Section 6.4, “Main Control Room Habitability Systems.”
2. Section 9.4.1, “Nuclear Island Non-Radioactive Ventilation System.”
3. SECY-95-132, “Policy and Technical Issues Associated With The Regulatory Treatment of Non-Safety Systems (RTNSS) In Passive Plant Designs (SECY-94-084),” May 22, 1995.
4. ASHRAE Standard 62-1989, “Ventilation for Acceptable Indoor Air Quality.”
5. Regulatory Guide 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,” Revision 1, December 2001.

B 3.7 PLANT SYSTEMS

B 3.7.7 Startup Feedwater Isolation and Control Valves

BASES

BACKGROUND

The startup feedwater system supplies feedwater to the steam generators during plant startup, hot standby and cooldown, and in the event of main feedwater unavailability.

The startup feedwater system serves no safety related function and has no safety related design basis, except to isolate feedwater in the event of a feedwater, steam line break, a steam generator tube rupture or other secondary side event.

The startup feedwater system consists of a flow path to each of the steam generators. Each flow path consists of two series startup feedwater valves to provide feedwater control for low feedwater demand conditions. Feedwater can be supplied to the startup feedwater line via either the main or startup feedwater pumps. The feedwater is delivered directly to the SG independent of the main feedwater line. Each startup feedwater line contains one control valve and one isolation valve (Ref. 1).

APPLICABLE SAFETY ANALYSES

The basis for the requirement to isolate the startup feedwater system is established by the analysis for large Steam Line Break (SLB) inside containment. It is also based on the analysis for a large Feedline Break (FLB) and a steam generator tube rupture.

Failure to isolate the startup feedwater system following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment. Failure to isolate the startup feedwater following a steam generator tube rupture may result in overfilling the steam generator.

Low T_{cold} or high steam generator level signals close the startup feedwater control and isolation valves and trips the startup feedwater pumps.

The startup feedwater isolation and control valves are components which actuate to mitigate a Design Basis Accident, and as such meet Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

This LCO ensures that the startup feedwater isolation and control valves will actuate on command, following a SLB, FLB or SGTR, and isolate startup feedwater flow to the steam generators.

The startup feedwater isolation and control valves are considered OPERABLE when they automatically close on an isolation actuation signal, and their isolation times are within the required limits.

APPLICABILITY

The startup feedwater isolation and control valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and the steam generators. In MODES 1, 2, 3 and 4, the startup feedwater isolation and control valves are required to be OPERABLE in order to limit the amount of mass and energy that could be added to containment in the event of a SLB or FLB and prevent steam generator overfill in the event of an SGTR. When the valves are closed, they are already performing their safety function.

In MODES 5 and 6, the energy in the steam generators is low, and isolation of the startup feedwater system is not required.

ACTIONS

The ACTIONS are modified by a Note allowing flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the flow paths can be rapidly isolated.

The second Note allows separate Condition entry for each flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable flow path.

<u>A.1 and A.2</u>

With only one isolation or control valve OPERABLE in one or more flow paths, there is no redundant capability to isolate the flow paths. In this case, both an isolation and a control valve in each flow path must be restored to OPERABLE status with 72 hours, or the flow path must be isolated. A Completion Time of 72 hours is acceptable since, with one valve in a flow path inoperable, there is a second valve available in the flow path to isolate the line.

BASES

ACTIONS (continued)

If the inoperable valve in the flow path can not be restored to OPERABLE status, then the flow path must be isolated within a Completion Time of 72 hours. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure.

For flow paths isolated in accordance with Required Action A.2.1, the affected flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that flow paths required to be isolated following an accident 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, through a system walkdown, that the isolation devices are in the correct position. The Completion Time of “once per 7 days” is appropriate considering the fact that the devices are operated under administrative controls, valve status indications in the main control room and the probability of their misalignment is low.

B.1

With both the isolation and control valves inoperable in one flow path, the affected flow path must be restored to OPERABLE status or isolated within a Completion Time of 8 hours. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure.

C.1, C.2, and C.3

If the isolation and control valves cannot be restored to OPERABLE status, closed, or isolated within the associated Completion Times, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in least MODE 3 within 6 hours, and in MODE 4 with RCS cooling provided by the normal residual heat removal system within 24 hours, and the affected flow path isolated 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.

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.7.7.1</u> This surveillance requires verification in accordance with the Inservice Testing Program to assure that both startup feedwater isolation and control valves are OPERABLE. The Surveillance Frequency is provided in the Inservice Testing Program.
REFERENCES	1. Section 10.4.9, "Startup Feedwater System."

B 3.7 PLANT SYSTEMS

B 3.7.8 Main Steam Line Leakage

BASES

BACKGROUND

A limit on leakage from the main steam line inside containment is required to limit system operation in the presence of excessive leakage. Leakage is limited to an amount which would not compromise safety consistent with the Leak-Before-Break (LBB) analysis discussed in Chapter 3 (Ref. 1). This leakage limit ensures appropriate action can be taken before the integrity of the lines is impaired.

LBB is an argument which allows elimination of design for dynamic load effects of postulated pipe breaks. The fundamental premise of LBB is that the materials used in nuclear plant piping are strong enough that even a large throughwall crack leaking well in excess of rates detectable by present leak detection systems would remain stable, and would not result in a double-ended guillotine break under maximum loading conditions. The benefit of LBB is the elimination of pipe whip restraints, jet impingement effects, subcompartment pressurization, and internal system blowdown loads.

As described in Section 3.6 (Ref. 1), LBB has been applied to the main steam line pipe runs inside containment. Hence, the potential safety significance of secondary side leaks inside containment requires detection and monitoring of leakage inside containment. This LCO protects the main steam lines inside containment against degradation, and helps assure that serious leaks will not develop. The consequences of violating this LCO include the possibility of further degradation of the main steam lines, which may lead to pipe break.

APPLICABLE SAFETY ANALYSES

The safety significance of plant leakage inside containment varies depending on its source, rate, and duration. Therefore, detection and monitoring of plant leakage inside containment are necessary. This is accomplished via the instrumentation required by LCO 3.4.9, “RCS Leakage Detection Instrumentation,” and the RCS water inventory balance (SR 3.4.7.1). Subtracting RCS leakage as well as any other identified non-RCS leakage into the containment area from the total plant leakage inside containment provides qualitative information to the operators regarding possible main steam line leakage. This allows the operators to take corrective action should leakage occur which is detrimental to the safety of the facility and/or the public.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Although the main steam line leakage limit is not required by the 10 CFR 50.36(c)(2)(ii) criteria, this specification has been included in Technical Specifications in accordance with NRC direction (Ref. 2).

LCO

Main steam line leakage is defined as leakage inside containment in any portion of the two (2) main steam line pipe walls. Up to 0.5 gpm of leakage is allowable because it is below the leak rate for LBB analyzed cases of a main steam line crack twice as long as a crack leaking at ten (10) times the detectable leak rate under normal operating load conditions. Violation of this LCO could result in continued degradation of the main steam line.

APPLICABILITY

Because of elevated main steam system temperatures and pressures, the potential for main steam line leakage is greatest in MODES 1, 2, 3, and 4.

In MODES 5 and 6, a main steam line leakage limit is not provided because the main steam system pressure is far lower, resulting in lower stresses and a reduced potential for leakage. In addition, the steam generators are not the primary method of RCS heat removal in MODES 5 and 6.

ACTIONS

A.1 and A.2

With main steam line leakage in excess of the LCO limit, the unit must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. The reactor must be placed in MODE 3 with 6 hours and MODE 5 within 36 hours. This action reduces the main steam line pressure and leakage, and also reduces the factors which tend to degrade the main steam lines. The Completion Time of 6 hours to reach MODE 3 from full power without challenging plant systems is reasonable based on operating experience. Similarly, the Completion Time of 36 hours to reach MODE 5 without challenging plant systems is also reasonable based on operating experience. In MODE 5, the pressure stresses acting on the main steam line are much lower, and further deterioration of the main steam line is less likely.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying that main steam line leakage is within the LCO limit assures the integrity of those lines inside containment is maintained. An early warning of main steam line leakage is provided by the automatic system which monitor the containment sump level. Main steam line leakage would appear as unidentified leakage inside containment via this system, and can only be positively identified by inspection. However, by performance of an RCS water inventory balance (SR 3.4.7.1) and evaluation of the cooling and chilled water systems inside containment, determination of whether the main steam line is a potential source of unidentified leakage inside containment is possible.

REFERENCES

1. Section 3.6, “Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping.”

2. NRC letter, Diane T. Jackson to Westinghouse (Nicholas J. Liparulo), dated September 5, 1996, “Staff Update to Draft Safety Evaluation Report (DSER) Open Items (OIs) Regarding the Westinghouse AP600 Advanced Reactor Design,” Open Item #365.

B 3.7 PLANT SYSTEMS

B 3.7.9 Fuel Storage Pool Makeup Water Sources

BASES

BACKGROUND

The spent fuel storage pool is normally cooled by the nonsafety spent fuel pool cooling system. In the event the normal cooling system is unavailable, the spent fuel storage pool can be cooled by the normal residual heat removal system. Alternatively, the spent fuel storage pool contains sufficient water inventory for decay heat removal by boiling. To support extended periods of loss of normal pool cooling, makeup water is required to provide additional cooling by boiling. Both safety and non-safety makeup water sources are available on-site.

Two safety-related, gravity fed sources of makeup water are provided to the spent fuel storage pool. These makeup water sources contain sufficient water to maintain spent fuel storage pool cooling for 72 hours. The containment cooling system water storage tank provides makeup water when pool decay heat is > 5.4 MWt and the decay heat in the reactor is less than 9.0 MWt. The cask washdown pit provides makeup water when decay heat in the pool is ≥ 4.6 MWt and ≤ 5.4 MWt. Additional on-site makeup water sources are available to provide fuel pool cooling between 3 and 7 days.

The containment cooling system water storage tank is isolated by two normally closed valves. The normally closed valves will be opened only to provide emergency makeup to the spent fuel storage pool. A third downstream valve permits the operator to regulate addition of water to the spent fuel storage pool as required to maintain the cooling water inventory.

Once decay heat in the fuel pool is reduced to below 4.6 MWt, the spent fuel storage pool water inventory is sufficient, without makeup, to maintain spent fuel storage pool for 72 hours. When the spent fuel storage pool decay heat load is reduced below 4.6 MWt, the cask washdown pit may be drained and returned to use for shipping cask cleaning operations.

A general description of the fuel storage pool design is given in Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in Section 9.1.3 (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES

In the event the normal spent fuel storage pool cooling system is unavailable, the spent fuel cooling is provided by the heat capacity of the water in the pool. The worst case decay heat load (decay heat > 5.4 MWt) is produced by an emergency full core off-load following a refueling plus ten years of spent fuel. For this case the spent fuel storage pool inventory provided by the water over the stored fuel and below the pump suction connection is capable of cooling the spent fuel storage pool without boiling for at least 2.5 hours, following a loss of normal spent fuel storage pool cooling. After boiling starts, makeup water may be required to replace water lost by boiling and is available, without offsite support, via the passive containment cooling water storage tank.

The requirements of LCO 3.6.6, "Passive Containment Cooling System – Operating," are applicable in MODES 1, 2, 3, and 4 and LCO 3.6.7, "Passive Containment Cooling System – Shutdown," are applicable in MODES 5 and 6 with decay heat > 9.0 MWt. LCOs 3.6.6 and 3.6.7 require availability of the containment cooling water tank for containment heat removal. Below 9.0 MWt decay heat, containment air cooling is adequate. Since there are no design conditions which result in both reactor decay heat > 9.0 MWt and spent fuel storage pool decay heat > 5.4 MWt, the applicability for LCOs 3.6.6/3.6.7 and for LCO 3.7.9 are mutually exclusive.

Since none of the Chapter 15 Design Basis Accident analyses assume availability of the containment cooling water tank or the cask washdown pit for spent fuel storage pool makeup, the fuel storage pool makeup water sources specification does not satisfy any of the 10 CFR 50.36(c)(2)(ii) criteria. This LCO is included in accordance with NRC guidance provided in an NRC letter (Reference 3).

LCO

The fuel storage pool makeup water sources, the cask washdown pit, and the containment cooling water tank are required to contain 13.75 ft and 400,000 gallons of water, respectively. An OPERABLE flow path from the required makeup source assures spent fuel cooling for at least 72 hours. Several additional makeup sources are available, including the ground level containment cooling ancillary water storage tank. These makeup sources assure spent fuel cooling for at least 7 days.

Note 1 specifies that either the cask washdown pit or the passive containment cooling water storage tank is required to be OPERABLE when the spent fuel storage pool decay heat ≥ 4.6 MWt and ≤ 5.4 MWt. Note 2 specifies that the passive containment cooling water storage tank source is required to be OPERABLE when the spent fuel storage pool decay heat is > 5.4 MWt, which is normal following a full core off load. The larger makeup source is necessary for the higher decay heat load.

BASES

LCO (continued)

When a portion of the fuel is returned to the reactor vessel in preparation for startup, the pool decay heat is reduced to ≤ 5.4 MWt and makeup from the cask washdown pit is sufficient.

APPLICABILITY

This LCO applies during storage of fuel in the fuel storage pool with a calculated decay heat ≥ 4.6 MWt. With decay heat < 4.6 MWt, the assumed spent fuel storage pool water inventory (i.e., level below the pump suction connection to the pool) provides for 3 days of cooling without makeup.

ACTIONS

LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since spent fuel pool cooling requirements apply at all times, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. Spent fuel pool cooling requirements are 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.

LCO 3.0.8 is applicable while in MODE 5 or 6. Since spent fuel pool cooling requirements apply at all times, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. Spent fuel pool cooling requirements are independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

<u>A.1</u>

If the passive containment cooling water storage tank (with decay heat > 5.4 MWt) and/or the cask washdown pit (with decay heat ≥ 4.6 and ≤ 5.4 MWt) is inoperable, Action must be initiated immediately to restore the makeup source or its associated flow path to OPERABLE status.

Additionally, in order to provide the maximum cooling capability, the spent fuel pool should be filled to its maximum level. Nonsafety related makeup sources can be used to fill the pool. This action is not specified in the specification, since the benefit of adding approximately 6 inches of water to the pool is less than a 5% improvement in cooling capability.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies sufficient passive containment cooling system water storage tank volume is available in the event of a loss of spent fuel cooling.

The 7 day Frequency is appropriate because the volume in the passive containment cooling system water storage tank is normally stable and water level changes are controlled by plant procedures.

SR 3.7.9.2

This SR verifies sufficient cask washdown pit water volume is available in the event of a loss of spent fuel cooling. The 13.75 ft level specified provides makeup water for stored fuel with decay heat ≥ 4.6 and ≤ 5.4 MWt.

The 30 day Frequency is appropriate because the cask washdown pit has only one drain line which is isolated by series manual valves which are only operated in accordance with plant procedures, thus providing assurance that inadvertent level reduction is not likely.

SR 3.7.9.3

This SR requires verification of the OPERABILITY of the manual makeup water source isolation valves in accordance with the requirements and Frequency specified in the Inservice Testing Program. Manual valves PCS-PL-V009, PCS-PL-V045, PCS-PL-V051, isolate the makeup flow path from the passive containment cooling system water storage tank. Manual valves SFS-PL-V042, SFS-PL-V045, SFS-PL-V049, SFS-PL-V066 and SFS-PL-V068 isolate the makeup flow path from the cask washdown pit.

REFERENCES

1. Section 9.1.2, “Spent Fuel Storage.”

2. Section 9.1.3, “Spent Fuel Pool Cooling System.”

3. NRC letter, William C. Huffman to Westinghouse Electric Corporation, “Summary of Telephone Conference with Westinghouse to Discuss Proposed Design Changes to the AP600 Main Control Room Habitability System,” dated September 11, 1997.

B 3.7 PLANT SYSTEMS

B 3.7.10 Steam Generator Isolation Valves

BASES

BACKGROUND

The steam generator isolation valves consist of the power operated relief valve (PORV) block valves (SGS-PL-V027A & B), PORVs (SGS-PL-V233A & B), and blowdown isolation valves (SGS-PL-V074A & B and SGS-PL-V075A & B). The PORV flow paths must be isolated following a Steam Generator Tube Rupture (SGTR) to minimize radiological releases. The blowdown flow path must be isolated following Loss of Feedwater and Feedwater Line Break events to retain the steam generator water inventory for Reactor Coolant System (RCS) heat removal.

A PORV is installed in a 6 inch branch line off of the main steam line piping from each steam generator, to provide for controlled removal of reactor decay heat during normal reactor cooldown when the main steam isolation valves are closed or the turbine bypass system is not available. A normally-open block valve is provided in each PORV line to provide backup isolation capability. Both the PORV and the block valve receive a Protection and Safety Monitoring System (PMS) isolation signal on low steam line pressure. The block valve is also a containment isolation valve.

The blowdown line from each steam generator is provided with two series isolation valves, both located outside, but close to, containment. The blowdown valves receive a PMS isolation signal on low SG level and on PRHR actuation. The first blowdown isolation valve outside of containment is also a containment isolation valve.

The steam generator PORVs and the blowdown isolation valves fail closed on loss of control or actuation power. The steam generator PORV block valves fail as-is on loss of control or actuation power. The steam generator isolation valves may also be actuated manually.

Descriptions of the PORVs and SG blowdown isolation are found in Section 10.3.2.2.3 and Section 10.4.8 (Refs. 1 & 2).

APPLICABLE SAFETY ANALYSES

The PORV flow paths must be isolated following an SGTR to minimize radiological releases from the ruptured steam generator into the atmosphere. The PORV flow path is assumed to open due to high secondary side pressure, during the SGTR. Dose analyses take credit for subsequent isolation of the PORV flow path by the PORV and/or the block valve which receive a close signal on low steam line pressure.

APPLICABLE SAFETY ANALYSES (continued)

The blowdown flow path on each SG must be isolated following Loss of Feedwater and Feedwater Line Break events to retain the steam generator water inventory for use in Reactor Coolant System (RCS) heat removal via the SGs. RCS heat removal for these events is, primarily, provided by the Passive Residual Heat Removal Heat Exchanger (PRHR HX); however, the SG heat removal is assumed. The SG blowdown isolation valves receive an isolation signal on low SG level or PRHR actuation. These events take credit for steam generator heat removal using the water inventory retained after blowdown isolation. If the blowdown line were not isolated, much of the inventory would drain from the SG rather than cool the RCS.

The steam generator isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the steam generator isolation valves consisting of the PORV, PORV block valve, and blowdown isolation valves on each steam generator to be OPERABLE. These isolation valves are considered OPERABLE when the valves are capable of closing on a PMS actuation signal.

This LCO provides assurance that the PORV and PORV block valve will perform their design safety function to mitigate the consequences of an SGTR that could result in offsite exposures.

Additionally, this LCO provides assurance that the steam generator blowdown isolation valves will perform their design safety function to mitigate the consequences of Loss of Feedwater and Feedwater Line Break events by retaining the steam generator water inventory for Reactor Coolant System (RCS) heat removal.

APPLICABILITY

The steam generator isolation valves must be OPERABLE in MODES 1, 2, and 3, and in MODE 4 with the RCS cooling not being provided by the Normal Residual Heat Removal System (RNS).

In MODE 4 with the RCS cooling being provided by the RNS and in MODES 5 and 6, the steam generators are not needed for RCS cooling and the potential for an SGTR or Loss of Feedwater and Feedwater Line Break events is minimized due to the reduced mass and energy in the RCS and steam generators.

ACTIONS

The ACTIONS are modified by a Note allowing the blowdown isolation 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 flow path can be rapidly isolated when a need for blowdown isolation is indicated.

The second Note allows separate Condition entry for each steam generator isolation flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable flow path.

A.1

With one valve in one or more PORV flow paths inoperable, action must be taken to isolate the flow path with a closed and deactivated valve. The valve must be deactivated to assure that the flow path will not be opened by a high pressure signal during the course of an SGTR event. This action places the flow path in a condition which assures the safety function is performed. A Completion Time of 72 hours is based on the availability of one OPERABLE PORV flow path isolation valve which is fully capable of performing the required isolation function.

B.1 and B.2

With one valve in one or more blowdown flow paths inoperable, action must be taken to isolate the flow path with a closed valve. This action places the flow path in a condition which assures the safety function is performed. A Completion Time of 72 hours to isolate the flow path is based on the availability of one OPERABLE blowdown flow path isolation valve which is fully capable of performing the required isolation function.

Since the blowdown isolation valve is not deactivated, periodic verification is required to assure that the flow path remains isolated. 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 status indications available in the control room, and other administrative controls, to ensure that the valve remains in the closed position.

C.1

With both valves in one or more PORV flow paths inoperable, action must be taken to isolate the flow path with a closed and deactivated valve. The valve must be deactivated to assure that the flow path will not be opened by a high pressure signal during the course of an SGTR event. This

ACTIONS (continued)

action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG isolation valves. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk in Reference 6.

D.1 and D.2

With two valves in one or more blowdown flow paths inoperable, action must be taken to isolate the flow path with a closed valve. This action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG isolation valves. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk in Reference 3.

Since the blowdown isolation valve is not deactivated, periodic verification is required to assure that the flow path remains isolated. 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 status indications available in the control room, and other administrative controls, to ensure that the valve remains in the closed position.

E.1 and E.2

If the SG isolation valves 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 with the RCS cooling provided by the RNS within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

The function of the SG isolation valves (PORV block valves (SGS-PL-V027A & B), PORVs (SGS-PL-V233A & B) and blowdown isolation valves (SGS-PL-V074A & B and SGS-PL-V075A & B)) is to isolate the steam generators in the event of SGTR, Loss of Feedwater or Feedwater Line Break. Stroking the valves closed demonstrates their capability to perform the isolation function. The Frequency for this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. Section 10.3.2.2.3, “Power-Operated Atmospheric Relief Valves.”

2. Section 10.4.8, “Steam Generator Blowdown System.”

3. Regulatory Guide 1.177, 8/98, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications.”

B 3.7 PLANT SYSTEMS

B 3.7.11 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 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 (Ref. 1). The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 2) 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 3.7.12, “Spent Fuel Pool Storage.” Prior to movement of an assembly, it is necessary to perform SR 3.7.12.1.

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 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 758 ppm established by critically analysis (Ref. 3). Compliance with the LCO minimum boron concentration limit of 2300 ppm ensures that the credited concentration is always available.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The fuel storage pool boron concentration is required to be ≥ 2300 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in References 1 and 3. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool.

ACTIONS

LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since spent fuel pool cooling requirements apply in all MODES when fuel is stored in the spent fuel storage pool, the ACTIONS have been modified by the Note stating that LCO 3.0.3 is not applicable. Spent fuel pool boron concentration requirements are 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.

LCO 3.0.8 is applicable while in MODE 5 or 6. Since spent fuel pool cooling requirements in all MODES when fuel is stored in the spent fuel storage pool, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. Spent fuel pool boron concentration requirements are independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1, A.2.1, and A.2.2

When the concentration of boron in the 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 of boron is restored simultaneously with suspending movement of fuel assemblies. An acceptable alternative is to verify by administrative means that the fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.11.1 This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is 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.
REFERENCES	1. AP1000 Design Control Document, Rev. 15, Sections 9.1.2, "Spent Fuel Storage" and 15.7.4, "Fuel Handling Accident." 2. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A). 3. APP-GW-GLR-029, "AP1000 Spent Fuel Storage Racks Critically Analysis," June 2006.

B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool Storage

BASES

BACKGROUND

The high density spent fuel storage racks are divided into two separate and distinct regions and include locations for storage of defective fuel as shown in Figure 4.3-1. Region 1, with a maximum of 243 storage locations and the Defective Fuel Cells, with 5 storage locations are 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, burnup, and decay time. Region 2, with a maximum of 641 storage locations is designed to accommodate spent fuel assemblies in all locations which comply with the combination of initial enrichment, burnup and decay time limits specified in LCO Figure 3.7.12-1, Fuel Assembly Burnup Requirements for the Region 2, “All Cell” Storage Configuration. Use of the IFE fuel rod storage canister is subject to the same storage requirements as the fuel assemblies.

Additionally, a second scheme “1-out-of-4 5.0 weight-percent fresh” is available for Region 2 as shown in Figure 4.3-2. New 5.0 weight percent U-235 fuel or any spent fuel may be stored in one location. Spent fuel (equivalent to 1.361 new fuel) shall be stored in the other three locations. The combination of initial enrichment, burnup, and decay time of the three spent fuel assemblies shall comply with the limits specified in LCO Figure 3.7.12-2, Fuel Assembly Burnup Requirements for the Region 2 “1-out-of-4 5.0 weight-percent fresh” Storage Configuration. The set of four relative storage locations may be repeated throughout Region 2. If the “1-out-of-4 5.0 weight-percent fresh” and the “All Cell” configurations are used together, the fuel in the storage locations surrounding the “1-out-of-4 5.0 weight-percent fresh” group(s) shall meet the LCO 3.7.12, Figure 3.7.12-2 limits.

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.

BACKGROUND (continued)

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 an inadvertent misplacement of a new fuel assembly. This accident has the potential for more than negligible positive reactivity effect 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, 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, burnup and decay time of the stored fuel in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.12.1.

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, “Fuel Storage 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 fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(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 758 ppm.

“All Cell” Storage Configuration

The “All Cell” storage configuration permits storage in all Region 2 locations of spent fuel which meets the combination of initial enrichment, burnup and decay time requirements shown in LCO Figure 3.7.12-1, Fuel Assembly Burnup Requirements for the Region 2, “All Cell” Storage

BASES

LCO (continued)

Configuration. Figure 3.7.12-1 permits new (no burnup) 1.627 weight percent U-235 fuel to be stored in the All Cell configuration. "1-out-of-4 5.0 weight-percent fresh" Storage Configuration Fuel stored in accordance with the "1-out-of-4 5.0 weight-percent fresh" storage configuration shall be stored in the relative locations shown in Figure 4.3-2. The "1-out-of-4 5.0 weight-percent fresh" storage configuration permits storage of 5.0 weight percent U-235 new (no burnup) fuel or any spent fuel in one specified location, provided fuel stored in the three remaining locations meets the enrichment, burnup and decay time requirements shown in LCO Figure 3.7.12-2, Fuel Assembly Burnup Requirements for the Region 2 "1-out-of-4 5.0 w/o Fresh" Storage Configuration. Figure 3.7.12-2 permits new (no burnup) 1.361 weight percent U-235 fuel to be stored in the three remaining locations. The 4-location configuration may be repeated throughout Region 2.

Interface Requirements

Fuel may be stored in both the "All Cell" and "1-out-of-4 5.0 weight-percent fresh" configurations at the same time, provided fuel stored in the interface locations around the "1-out-of-4 5.0 weight-percent fresh" configuration group(s) meets the LCO Figure 3.7.12-1 requirements. Fuel assemblies not meeting the criteria of Figures 3.7.12-1 and 3.7.12-2 shall be stored in accordance with Specification 4.3.1.1.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 of this fuel storage pool.

ACTIONS

LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since spent fuel pool storage requirements apply in all MODES when fuel is stored in Region 2 or 3, the ACTIONS have been modified by a Note stating the LCO 3.0.3 is not applicable. Spent fuel pool storage requirements are 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.

LCO 3.0.8 is applicable while in MODE 5 or 6. Since spent fuel pool storage requirements apply in all MODES when fuel is stored in Region 2 or 3, the ACTIONS have been modified by a Note stating the LCO 3.0.8 is not applicable. Spent fuel pool storage requirements are independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

BASES

ACTIONS (continued)

A.1

The LCO is not met if spent fuel assemblies stored in Region 2 “All Cell,” “1-out-of-4 5.0 weight-percent fresh” or interface spent fuel assembly storage locations do not meet the applicable initial enrichment, burnup and decay time limits in accordance with Figure 3.7.12-1 or 3.7.12-2.

Additionally, LCO is not met if fuel, required to be stored in the New Fuel location of the “1-out-of-4 5.0 weight-percent fresh” storage configuration, is misplaced. When the LCO is not met, action must be initiated immediately to make the necessary fuel assembly movement(s) in Region 2 to bring the storage configuration into compliance with Figures 3.7.12-1 and 3.7.12-2 or to move fuel to Region 1 or the defective fuel cells.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies by administrative means that the initial enrichment, burnup and decay time of the fuel assembly is in accordance with Figure 3.7.12-1 or 3.7.12-2 as applicable for “All Cell,” “1-out-of-4 5.0 weight-percent fresh” and interface spent fuel assembly storage locations. Fuel stored in Region 2 that does not meet the Figure 3.7.12-1 or 3.7.12-2 limits shall be stored in Figure 4.3-1 “1-out-of-4 5.0 weight-percent fresh” New Fuel location.

REFERENCES

1. Double contingency principle ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. APP-GW-GLR-029, “AP1000 Spent Fuel Storage Racks Criticality Analysis,” June 2006.

3. AP1000 Design Control Document, Rev. 15, Sections 9.12, “Spent Fuel Storage” and 15.7.4, “Fuel Handling Accident.”

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 DC Sources – Operating

BASES

BACKGROUND

The Class 1E DC and UPS System (IDS) provides electrical power for safety related and vital control instrumentation loads, including monitoring and main control room emergency lighting. It also provides power for safe shutdown when all the onsite and offsite AC power sources are lost and cannot be recovered for 72 hours. As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the Class 1E DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The Class 1E DC electrical power system also conforms to the requirements of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 250 VDC electrical power system consists of four independent safety related Class 1E DC electrical power subsystems (Division A, B, C, and D). Divisions A and D each consist of one 24 hour battery bank, one battery charger, and the associated control equipment and interconnecting cable. Divisions B and C each consist of two battery banks (one 24 hour and one 72 hour), two battery chargers, and the associated control equipment and interconnecting cabling. The loads on the battery banks (including those on the associated inverters) are grouped according to their role in response to a Design Basis Accident (DBA). Loads which are a one time or limited duration load (engineered safeguards features (ESF) actuation cabinets and reactor trip function) required within the first 24 hours following an accident are connected to the “24 hour” battery bank. Loads which are continuous or required beyond the first 24 hours following an accident (emergency lighting, post accident monitoring, and Qualified Data Processing System) are connected to the “72 hour” battery bank. There are a total of six battery banks. A battery bank consists of two battery strings connected in series. Each battery string consists of 60 cells connected in series. Divisions A and D each have one 2400 ampere hour battery bank and Divisions B and C each have two 2400 ampere hour battery banks.

Additionally, there is one installed spare battery bank and one installed spare battery charger, which provide backup service in the event that one of the battery banks and/or one of the preferred battery chargers is out of service. The spare battery bank and charger are Class 1E and have the same rating as the primary components. If the spare battery bank with the charger is substituted for one of the preferred battery banks or chargers, then the requirements of independence and redundancy between subsystems are maintained and the division is OPERABLE.

BASES

BACKGROUND (continued)

During normal operation, the 250 VDC load is powered from the battery chargers with the batteries 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.

Each battery bank provides power to an inverter, which in turn powers an AC instrumentation and control bus. The AC instrumentation and control bus loads are connected to inverters according to the battery bank type, 24 hour or 72 hour.

The Class 1E DC power distribution system is described in more detail in Bases for LCO 3.8.5, "Distribution System – Operating," and LCO 3.8.6, "Distribution System – Shutdown."

Each battery has adequate storage capacity to carry the required load for the required duration as discussed in Reference 4.

Each 250 VDC battery bank, including the spare battery bank, is separately housed in a ventilated room apart from its charger and distribution centers. 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 separate subsystem. There is no sharing between separate Class 1E subsystems such as batteries, battery chargers, or distribution panels.

The batteries for each Class 1E electrical power subsystem are based on 125% of required capacity. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 256 V per battery discussed in Reference 4. The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

Each electrical power subsystem 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 has sufficient capacity to restore the battery bank from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads (Ref. 4).

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the Chapter 6 (Ref. 6) and Chapter 15, (Ref. 7), assume that engineered safety features are OPERABLE. The Class 1E DC electrical power system provides 250 volts power for safety related and vital control instrumentation loads

APPLICABLE SAFETY ANALYSES (continued)

including monitoring and main control room emergency lighting during all MODES of operation. It also provides power for safe shutdown when all the onsite and offsite AC power sources are lost.

The OPERABILITY of the Class 1E DC sources is consistent with the initial assumptions of the accident analyses. This includes maintaining at least three of the four divisions of DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite and onsite AC power sources; and

b. A worst case single failure.

The DC Sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Class 1E DC electrical power subsystems 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 anticipated operational occurrence (AOO) or a postulated DBA. Loss of Class 1E DC electrical power from one division does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE Class 1E DC electrical power subsystem requires all required batteries and respective chargers to be operating and connected to the associated DC bus(es). The spare battery and/or charger may be used by one subsystem for OPERABILITY.

APPLICABILITY

The Class 1E 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 DBA.

Class 1E DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.2, “DC Sources – Shutdown.”

BASES

ACTIONS

A.1, A.2, and A.3

Condition A represents one division with one or two battery chargers inoperable (e.g., the voltage limit of SR 3.8.1.1 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 6 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 24 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.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

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 24 hours, avoiding a premature shutdown with its own attendant risk.

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 6 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.

ACTIONS (continued)

If the charger is operating in the current limit mode after 6 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 24 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 24 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 7 days. 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 7 day Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

B.1, B.2, and B.3

Condition B represents two divisions with one or more battery chargers inoperable (e.g., the voltage limit of SR 3.8.1.1 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 B.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 24 hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to the charger inoperability.

ACTIONS (continued)

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 24 hours, avoiding a premature shutdown with its own attendant risk.

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 24 hours (Required Action B.2).

Required Action B.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 24 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 B.3 limits the restoration time for the inoperable battery charger to 7 days. 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 7 day Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

<u>C.1</u>

Condition C represents one division with one or more batteries inoperable. With one or more batteries inoperable, the DC bus is being supplied by the OPERABLE battery chargers. Any event that results in a loss of the AC bus supporting the battery chargers will also result in loss of DC to that train.

ACTIONS (continued)

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC electrical power subsystem; however, all applicable Surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

D.1

Condition D represents two divisions with one or more batteries inoperable. With one or more batteries inoperable, the DC bus is being supplied by the 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 train. 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 3.8.1, 3.8.2, and 3.8.7 together with additional specific completion times.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC electrical power subsystem; however, all applicable Surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

E.1

If one of the Class 1E DC electrical power subsystems is inoperable, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate all design basis accidents, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs

BASES

ACTIONS (continued)

following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System (PMS) (LCO 3.3.2, ESFAS Instrumentation). Depending on the nature of the DC electrical power subsystem inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS Division is similar to loss of one DC electrical power subsystem. In both cases, actuation of the safety functions associated with one of the four subsystems/divisions may no longer be available.

F.1

Condition F represents two subsystems 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 distribution subsystems. The 2 hour limit is consistent with the allowed time for two inoperable DC distribution subsystems.

If two of the required DC electrical power subsystems are inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the two remaining DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate all but the very worst case events. Since a subsequent worst case single failure would, however, result in the loss of the third subsystem, leaving only one subsystem with limited capacity to mitigate events, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 11) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

BASES

ACTIONS (continued)

G.1 and G.2

If the inoperable DC electrical power 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 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 unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.1.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the battery chargers which support 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 initial voltages assumed in the battery sizing calculations. 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 3.8.1.2

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 ensure that these requirements can be satisfied.

This SR provides two options. One option requires that each battery charger be capable of supplying 200 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

SURVEILLANCE REQUIREMENTS (continued)

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, an the exponential decay in charging current. The battery is recharged when the measured charging current is ≤ 2 amps.

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.

<u>SR 3.8.1.3</u>

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the Class 1E DC electrical power system. The discharge rate and test length corresponds 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 with intervals between tests not to exceed 24 months. This Surveillance may be performed during any plant condition with the spare battery and charger providing power to the bus.

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 modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published 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 rated one minute discharge represents a very small portion of the battery capacity,

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 should 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.

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.

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems if the spare battery is not connected. This restriction from normally performing the Surveillance in MODE 1 or 2 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.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.

2. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," U.S. Nuclear Regulatory Commission, March 10, 1971.

BASES

REFERENCES (continued)

3. IEEE-308 1991, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.

4. Section 8.3.2, "Class 1E DC Power Systems."

5. IEEE-485 1997, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," Institute of Electrical and Electronic Engineers, June 1983.

6. Chapter 6, "Engineered Safety Features."

7. Chapter 15, "Accident Analyses."

8. IEEE-450 1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," Institute of Electrical and Electronic Engineers, June 1986.

9. Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1977.

10. Regulatory Guide 1.129 Revision 1, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1978.

11. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 DC Sources – Shutdown

BASES

BACKGROUND	A description of the Class 1E DC power sources is provided in the Bases for LCO 3.8.1, “DC Sources – Operating.”

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the emergency auxiliaries and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystem is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems’ OPERABILITY.

The OPERABILITY of the minimum Class 1E DC 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 Class 1E DC power sources are provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or 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 Design Basis Accidents (DBAs) 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

BASES

APPLICABLE SAFETY ANALYSES (continued)

consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

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 Design Basis 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, have 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 Class 1E DC Sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Class 1E DC electrical power subsystems are required to be OPERABLE to support required trains of Class 1E Distribution System divisions required to be OPERABLE by LCO 3.8.6. This ensures the availability of sufficient Class 1E DC 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, inadvertent reactor vessel draindown).

As described in the previous section, “Applicable Safety Analyses,” in the event of an accident during shutdown, the Technical Specifications are designed to maintain the plant in such a condition that, even with a single failure, the plant will not be in immediate difficulty.

BASES

APPLICABILITY

The Class 1E DC power 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 in case of an inadvertent draindown of the reactor vessel;

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

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The Class 1E DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1, "DC Sources – Operating."

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 and A.2

With one or more of the required (per LCO 3.8.6, "Distribution Systems – Shutdown") Class 1E DC power subsystems inoperable, the remaining subsystems may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS, fuel movement, and/or operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances this option would likely involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity

BASES

ACTIONS (continued)

additions that could result in failure to meet the minimum SDM or boron concentration limit) to assure continued safe operation. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

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 electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary Class 1E DC electrical power to the unit safety systems.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC power subsystem; however, all applicable surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required Class 1E DC electrical power 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 3.8.2.1

SR 3.8.2.1 requires performance of all Surveillances required by SR 3.8.1.1 through SR 3.8.1.3. Therefore, see the corresponding Bases for LCO 3.8.1 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 sources 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. Chapter 6, “Engineered Safety Features.”
2. Chapter 15, “Accident Analysis.”

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Inverters – Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the Class 1E AC instrument and control buses because of the stability and reliability they achieve. Divisions A and D, each consist of one Class 1E inverter. Divisions B and C, each consist of two inverters. The function of the inverter is to convert Class 1E DC electrical power to AC electrical power, thus providing an uninterruptible power source for the instrumentation and controls for the Protection and Safety Monitoring System (PMS). The inverters are powered from the Class 1E 250 V battery sources (Ref. 1).

Under normal operation, a Class 1E inverter supplies power to the Class 1E AC instrument and control bus. If the inverter is inoperable or the Class 1E 250 VDC input to the inverter is unavailable, the Class 1E AC instrument and control bus is powered from the backup source associated with the same division via a static transfer switch featuring a make-before-break contact arrangement. In addition, a manual mechanical bypass switch is used to provide a backup power source to the Class 1E AC instrument and control bus when the inverter is removed from service. The backup source is a Class 1E regulating 480-208/ 120 volt transformer providing a regulated output to the Class 1E AC instrument and control bus through a static transfer switch and a manual bypass switch.

In addition to powering safety loads, the Class 1E AC power sources are used for emergency lighting in the main control room and remote shutdown workstation. When a normal AC power source for emergency lighting is lost, the loads are automatically transferred to a Class 1E AC power source. Specific details on inverters and their operating characteristics are found in Chapter 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) transient analyses in Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume engineered safety features are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the PMS instrumentation and controls 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 Technical Specifications 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 at least three of the four Divisions of AC instrument and control buses OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite and onsite AC power source; and

b. A worst case single failure.

Inverters are a part of distribution systems, and as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the PMS instrumentation and controls is maintained. The six inverters ensure an uninterruptible supply of AC electrical power to the six Class 1E AC instrument and control buses even if all AC power sources are de-energized.

OPERABLE inverters require that the Class 1E AC instrument and control bus be powered by the inverter with output voltage and frequency within tolerances, and the power input to the inverter from a 250 VDC station battery.

This LCO is modified by a Note that allows one inverter to be disconnected from its associated Class 1E DC bus for ≤ 72 hours, if the associated Class 1E AC instrument and control bus is powered from its Class 1E regulating transformer during the period and all other inverters are OPERABLE. This allows an equalizing charge to be placed on one battery bank. If the inverter was not disconnected, the resulting voltage condition might damage the inverter. These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 72 hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected Class 1E AC instrument and control bus while taking into consideration the time required to perform an equalizing charge on the battery bank.

BASES

LCO (continued)

The intent of this Note is to limit the number of inverters that may be disconnected. Only the inverter associated with the single battery bank undergoing an equalizing charge may be disconnected. All other inverters must be aligned to their associated batteries.

APPLICABILITY

The 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 DBA.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.4, "Inverters – Shutdown."

ACTIONS

A.1

With a required inverter inoperable, its associated Class 1E AC instrument and control bus is automatically energized from its regulating transformer. A manual switch is also provided which can be used if the static transfer switch does not properly function.

For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.5, "Distribution System – Operating." This ensures that the vital bus is re-energized within 12 hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour time 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 instrument and control bus is powered from its regulating transformer, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC instrument and control buses is the preferred source for powering instrumentation trip setpoint devices.

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to MODE 5 where the probability and consequences on an event are minimized. 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 unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This Surveillance verifies that the inverters are functioning properly with all required switches and circuit breakers closed and Class 1E AC instrument and control buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the PMS instrumentation connected to the Class 1E AC instrument and control buses. The 7 day Frequency takes into account the effectiveness of the voltage and frequency instruments, the redundant capability of the inverters, and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. Section 8.3.2.1.1.2, “Class 1E Uninterruptible Power Supplies.”
2. Chapter 6, “Engineered Safety Features.”
3. Chapter 15, “Accident Analyses.”

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 Inverters – Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for Specification 3.8.3, "Inverters – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Protection and Monitoring System Engineered Safety Feature Actuation System instrumentation and controls 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 minimum inverters to each Class 1E AC instrument and control bus during MODES 5 and 6, ensures that (Refs. 1 and 2):

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 an inadvertent draindown of the vessel or 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 Design Basis Accidents (DBAs) 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

BASES

APPLICABLE SAFETY ANALYSES (continued)

consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

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 Design Basis 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, have 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 Class 1E UPS inverters are part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The inverters ensure 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 postulated DBA. The battery powered inverters provide an uninterruptible supply of AC electrical power to the Class 1E AC instrument and control buses, even if the normal power supply from the 480 VAC is deenergized. OPERABILITY of the inverters requires that the Class 1E instrument and control buses be powered by the inverter with output voltage and frequency within tolerances, and the power input to the inverter from a 250 VDC station battery. This ensures the availability of sufficient inverter power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (fuel handling accidents, inadvertent reactor vessel draindown).

BASES

APPLICABILITY

The inverters 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 in case of an inadvertent draindown of the reactor vessel;

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.

Class 1E UPS inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.3, “Inverters – Operating.”

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.

<u>A.1 and A.2</u>

If one or more required (per LCO 3.8.6, Distribution Systems – Shutdown) inverters are inoperable, the remaining OPERABLE inverters may be capable of supporting required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By allowance of the option to declare required features inoperable with 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., suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, 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 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 inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter 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 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 regulating transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and Class 1E AC instrument and control 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 Class 1E AC instrument and control buses. The 7 day Frequency takes into account the effectiveness of the voltage and frequency instruments, the redundant capability of the inverters, and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. Chapter 6, “Engineered Safety Features.”

2. Chapter 15, “Accident Analysis.”

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 Distribution Systems – Operating

BASES

BACKGROUND

The onsite Class 1E and DC and UPS electrical power distribution system is divided by division into four independent AC and DC electrical power distribution subsystems (Divisions A, B, C, and D).

The Class 1E AC distribution Divisions A and D each consists of one 208/120 V bus. The Class 1E AC distribution Divisions B and C each consists of two 208/120 V buses. The buses are normally powered from separate inverters which are connected to the respective Division Class 1E battery banks. The backup source provided for each Division for the Class 1E AC instrument and control buses is a Class 1E regulating transformer providing regulated output to the Class 1E AC instrument and control buses through a static transfer switch and a manual bypass switch. Power to the transformer is provided by the nonsafety related Main AC Power System. Additional description of this system may be found in the Bases for Specification 3.8.3, "Inverters – Operating."

The Class 1E DC distribution Divisions A and D each consists of one 250 VDC bus. The Class 1E DC distribution Divisions B and C each consists of two 250 VDC buses. The buses for the four Divisions are normally powered from their associated Division battery chargers. The backup source for each Class 1E DC bus is its associated Class 1E battery bank. Additionally, there is one installed spare Class 1E battery bank and one installed spare Class 1E battery charger, which can provide backup power to a Class 1E DC bus in the event that one of the battery banks or one of the chargers is out of service. Additional description of this system may be found in the Bases for Specification 3.8.1, "DC Sources Operating."

The list of all required distribution buses is presented in Table B 3.8.5-1 and shown in Section 8.3.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume engineered safety features (ESFs) are OPERABLE. The Class 1E AC and DC electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the ESFs so that the fuel, Reactor Coolant System (RCS) and containment design limits are not exceeded.

APPLICABLE SAFETY ANALYSES (continued)

These limits are discussed in more detail in the Bases for Technical Specifications 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

The OPERABILITY of the Class 1E AC and DC 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 at least three of the four Divisions of Class 1E AC and DC power distribution systems OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite and onsite AC power sources; and

b. A worst case single failure.

The Class 1E AC and DC electrical power distribution system satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The required power distribution subsystems listed in Table B 3.8.5-1 ensure the availability of Class 1E AC and DC 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 DBA. The Division A, B, C, and D Class 1E AC and DC electrical power distribution subsystems are required to be OPERABLE.

Maintaining the Division A, B, C, and D AC and DC electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of the ESFs is not defeated. Three of the four Class 1E AC and DC power distribution subsystems are capable of providing the necessary electrical power to the associated ESF components. Therefore, a single failure within any subsystem or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

OPERABLE Class 1E DC electric power distribution subsystems require the associated buses, motor control centers, and electrical circuits to be energized to their proper voltage from either the associated battery bank or charger. The spare battery bank and/or chargers may be used by one subsystem for OPERABILITY. OPERABLE Class 1E AC electrical power distribution subsystems require the associated buses to be energized to their proper voltages and frequencies from the associated inverter or regulating transformer.

APPLICABILITY

The Class 1E AC and DC 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 DBA.

The Class 1E AC and DC electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for Specification 3.8.6, “Class 1E Distribution Systems – Shutdown.”

ACTIONS

A.1

With one division of the Class 1E AC instrument and control bus inoperable the remaining Class 1E AC instrument and control buses have the capacity to support a safe shutdown and to mitigate all DBAs, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E AC instrument and control buses have the capacity to support a safe shutdown and to mitigate most design basis accidents following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one AC instrument and control bus against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

This 6 hour limit is shorter than Completion Times allowed for most supported systems which 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 6 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;

ACTIONS (continued)

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 6 hour Completion Time takes into account the importance to safety of restoring the Class 1E AC instrument and control bus to OPERABLE status, the passive design of the ESF systems, the redundant capability afforded by the other OPERABLE Class 1E AC instrument and control buses, and the low probability of a DBA occurring during this period which requires more than two OPERABLE AC instrument and control buses.

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System division (LCO 3.3.2, ESFAS Instrumentation). Depending on the nature of the AC instrument and control inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS division is similar to loss of one division AC instrument and control bus. In both cases, actuation of the safety functions associated with one of the four subsystems/divisions may no longer be available.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 6 hours. This could lead to a total of 12 hours, since initial failure of the LCO, to restore the AC instrument and control distribution system. At this time, a DC circuit could again become inoperable, and AC instrument and control distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal “time zero” for beginning the allowed outage time “clock.” This will result in establishing the “time zero” at the time the LCO was initially not met, instead of the time Condition A was entered. The 12 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

ACTIONS (continued)

B.1

With one Division of the Class 1E DC electrical power distribution subsystem inoperable, the remaining Divisions have the capacity to support a safe shutdown and to mitigate all DBAs, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining Divisions have the capacity to support a safe shutdown and to mitigate most design basis accidents following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one Division against the risks of a forced shutdown. Additionally, the completion time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System division (LCO 3.3.2, ESFAS Instrumentation). Depending on the nature of the DC electrical power distribution subsystem inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS division is similar to loss of one DC electrical power distribution subsystem. In both cases, actuation of the safety functions associated with one of the four subsystems/divisions may no longer be available.

This 6 hour limit is shorter than Completion Times allowed for most supported systems which 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 6 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 to restore power to the affected division; and

ACTIONS (continued)

c. The potential for an event in conjunction with a single failure of a redundant component.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC instrument and control bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 6 hours. This could lead to a total of 6 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable, and DC distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal “time zero” for beginning the allowed outage time “clock.” This will result in establishing the “time zero” at the time the LCO was initially not met, instead of the time Condition B was entered. The 12 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With two divisions of AC instrument and control buses inoperable, the remaining OPERABLE 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 required ESF functions not being supported. Therefore, the required divisions of AC instrument and control buses 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 constant voltage transformer.

Condition C represents two divisions of AC instrument and control 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 noninterruptable 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 buses and restoring power to the affected buses.

ACTIONS (continued)

This 2 hour time limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate AC instrument and control power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, which 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 AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; 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 instrument and control buses to OPERABLE status, the redundant capability afforded by the other OPERABLE buses, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action C.1 establishes a limit on the maximum allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, a DC bus is inoperable and subsequently returned to OPERABLE, the LCO may already have been not met for up to 12 hours. This could lead to a total of 14 hours, since initial failure of the LCO, to restore the bus distribution system. At this time, a DC train could again become inoperable, and AC bus distribution restored to OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal “time zero” for beginning the allowed outage time “clock.” This will result in establishing the “time zero” at the time the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

ACTIONS (continued)

D.1

With two divisions of DC electrical power distribution subsystems inoperable, 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 the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the required DC buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

Condition D represents two subsystems 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 divisions.

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 divisions; 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).

ACTIONS (continued)

The second Completion Time for Required Action D.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition D is entered while, for instance, an AC instrument and control bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 12 hours. This could lead to a total of 14 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable, and DC distribution restored to OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

E.1 and E.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to MODE 5 where the probability and consequences on an event are minimized. 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 unit conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With two Divisions with inoperable distribution subsystems that result in a loss of safety function, adequate core cooling, containment OPERABILITY and other vital functions for DBA mitigation would be compromised, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

This Surveillance verifies that the Class 1E AC and DC electrical power distribution subsystems are functioning properly, with the required circuit breakers and switches properly aligned. The verification of proper voltage availability on the buses ensures that the required voltage is

SURVEILLANCE REQUIREMENTS (continued)

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 Class 1E AC and DC electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. Section 8.3.2, "DC Power Systems."
2. Chapter 6, "Engineering Safety Features."
3. Chapter 15, "Accident Analyses."

Table B 3.8.5-1 (page 1 of 1)
Class 1E AC and DC Electrical Power Distribution System

TYPE	VOLTAGE	DIVISION A*	DIVISION B*	DIVISION C*	DIVISION D*
DC Buses	250 Vdc	IDSA-DS-1	IDSB-DS-1 IDSB-DS-2	IDSC-DS-1 IDSC-DS-2	IDSD-DS-1
DC Distribution Panels	250 Vdc	IDSA-DD-1 IDSA-DK-1	IDSB-DD-1 IDSB-DK-1	IDSC-DD-1 IDSC-DK-1	IDSD-DD-1 IDSD-DK-1
AC Instrumentation and Control Buses	120 Vac	IDSA-EA-1	IDSB-EA-1 IDSB-EA-3	IDSC-EA-1 IDSC-EA-3	IDSD-EA-1

* Each Division of the AC and DC electrical power distribution systems is a subsystem.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Distribution Systems – Shutdown

BASES

BACKGROUND

A description of the Class 1E AC instrument and control bus and Class 1E DC electrical power distribution system is provided in the Bases for Specification 3.8.5, "Distribution System – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The Class 1E AC and DC electrical power sources and associated power distribution systems are designed to provide sufficient capacity, redundancy, and reliability to ensure the availability of necessary power to the ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the minimum Class 1E AC and DC electrical power sources and associated 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 an inadvertent draindown of the vessel or a fuel handling accident.

The Class 1E AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(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.

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 Class 1E AC and DC electrical power distribution subsystems are 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 in case of an inadvertent draindown of the reactor vessel;

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 Class 1E AC and DC electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.5, “Distribution Systems – Operating.”

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.

<u>A.1 and A.2</u>

If one or more required Class 1E DC or Class 1E AC instrument and control bus electrical power distribution subsystems are inoperable, the remaining OPERABLE divisions may be capable of supporting required features to allow continuation of CORE ALTERATIONS, fuel movement, and/or operations with a potential for draining the reactor vessel. By

ACTIONS (continued)

allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions will be implemented in accordance with the affected equipment LCO Required Actions. In many instances this would likely involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, 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 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 will minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC 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.

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 3.8.6.1

This Surveillance verifies that the Class 1E AC and DC electrical power distribution subsystems are functioning properly, with the required circuit breakers and switches properly aligned. 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

SURVEILLANCE REQUIREMENTS (continued)

loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the electrical power distribution subsystems and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. Chapter 6, "Engineered Safety Features."

2. Chapter 15, "Accident Analysis."

3. Section 8.3.2, "DC Power Systems."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Battery Parameters

BASES

BACKGROUND

LCO 3.8.7, Battery Parameters, delineates the limits on electrolyte temperature, level, float voltage and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.1, “DC Sources – Operating,” and LCO 3.8.2, “DC Sources – Shutdown.” In addition to the limitations of this Specification, the licensee controlled program also implements a program specified in Specification 5.5.11 for monitoring various battery parameters that is based on the recommendations of IEEE Standard 450-1995, “IEEE Recommended Practice For Maintenance, Testing, And Replacement Of Vented Lead-Acid Batteries For Stationary Applications” (Ref. 3).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in Chapter 6 (Ref. 1), and Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for safety related and vital control instrumentation loads including monitoring and main control room emergency lighting during all MODES of operation. It also provides power for safe shutdown when all the onsite and offsite AC power sources are lost.

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 three of the four Divisions of DC sources OPERABLE during accident conditions, in the event of:

a. An assumed loss of all offsite and onsite AC power sources; and

b. A worst case single failure.

Battery parameters satisfy the Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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 DBA. Battery parameter limits are conservatively established, allowing continued DC electrical system function even with

BASES

LCO (continued)

limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the licensee controlled program is conducted as specified in Specification 5.5.11.

APPLICABILITY

The battery parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery parameter limits are only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.1, and LCO 3.8.2.

ACTIONS

A.1, A.2, and A.3

With one or more cells in one or more batteries in one Division < 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 3.8.1.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.7.1). 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 in one or more batteries < 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 3.8.1.1 or SR 3.8.7.1 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 3.8.7.1 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.

B.1 and B.2

One or more batteries in one Division with float > 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

ACTIONS (continued)

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 24 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 24 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 24 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 24 hours is a reasonable time prior to declaring the battery inoperable.

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.1.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

C.1, C.2, and C.3

With one or more batteries in one Division 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

ACTIONS (continued)

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 5.5.11, 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 5.5.11.b item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from Annex D of IEEE Standard 450-1995. 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 batteries may have to be declared inoperable and the affected cells replaced.

D.1

With one or more batteries in one Division 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.

E.1

With one or more batteries 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 redundant batteries are involved. With redundant 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 on non-redundant batteries not within limits are therefore not appropriate, and the parameters must be restored to within limits in three Divisions within 2 hours.

BASES

ACTIONS (continued)

F.1

With one or more batteries 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 in one Division 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 3.8.7.1

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. 3). The 7 day Frequency is consistent with IEEE-450 (Ref. 3).

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 3.8.1.1. When this float voltage is not maintained the Required Actions of LCO 3.8.1 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.

SR 3.8.7.2 and SR 3.8.7.5

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 264.0 V at the battery terminals, or 2.20 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 5.5.11. SRs 3.8.7.2 and 3.8.7.5 require verification that the cell float voltages are equal to or greater than the short term absolute

SURVEILLANCE REQUIREMENTS (continued)

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. 3).

SR 3.8.7.3

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. 3).

SR 3.8.7.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 60°F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provided 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. 3).

SR 3.8.7.6

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 3.8.7.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.1.3.

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

SURVEILLANCE REQUIREMENTS (continued)

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. 3) and IEEE-485 (Ref. 4). 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. 3), 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. 3).

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 or 2 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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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.

REFERENCES

1. Chapter 6, “Engineered Safety Features.”

2. Chapter 15, “Accident Analyses.”

3. IEEE-450 1995, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications.”

4. IEEE-485-1983, June 1983.

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentration of the Reactor Coolant System (RCS), the refueling cavity, and the transfer tube 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 procedures.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled down and depressurized, the vessel head is unbolted and slowly removed. The refueling cavity and the fuel transfer canal are then flooded with borated water from the In-containment Refueling Water Storage Tank (IRWST) by the use of the Spent Fuel Pool Cooling System (SFS).

During refueling, the water volumes in the RCS, the fuel transfer canal and the refueling cavity are contiguous. However, the soluble boron concentration is not necessarily the same in each volume. If additions of boron are required during refueling, the Chemical and Volume Control System (CVS) provides the borated makeup.

The pumping action of the Normal Residual Heat Removal System (RNS) in the RCS, the SFS pumps in the spent fuel pool and refueling cavity, and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the fuel transfer canal. The RNS is in operation during refueling to provide forced circulation in the RCS, while the SFS is in operation to cool and purify the spent fuel pool and refueling cavity. Their operation assists in maintaining the boron concentration in the RCS, the refueling cavity, and fuel transfer canal above the COLR limit.

BASES

APPLICABLE SAFETY ANALYSES	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. The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling cavity and the transfer tube while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} ≤ 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} of ≤ 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.
ACTIONS	<u>A.1 and A.2</u> Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the plant in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling cavity, or the fuel transfer canal is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately. Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude completion of actions to establish a safe condition, including moving a component to a safe position. <u>A.3</u> In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

BASES

ACTIONS (continued)

In determining the required combination of boration flow rate and concentration, no unique design basis accident (DBA) 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 shall begin boration with the best source available for plant operations.

Once boration is 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 verifies that the coolant boron concentration in the RCS, the refueling cavity and the fuel transfer canal is within the COLR limit. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is a sufficient interval to verify the boron concentration. The surveillance interval is based on operating experience, isolation of unborated water sources in accordance with LCO 3.9.2, and the availability of the source range neutron flux monitors required by LCO 3.9.3.

REFERENCES

1. Chapter 15, “Accident Analysis.”

2. NS-57.2, ANSI/ANS-57.2-1983, Section 6.4.2.2.3, American Nuclear Society, American National Standard, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants,” 1983.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Unborated Water Source Flow Paths

BASES

BACKGROUND	During MODE 6 operation, all flow paths for reactor makeup water sources containing unborated water which are connected to the Reactor Coolant System (RCS) must be closed to prevent an unplanned dilution of the reactor coolant. At least one isolation valve in each flow path must be secured in the closed position. The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition, made by reducing the boron concentration, is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution event.
APPLICABLE SAFETY ANALYSES	The possibility of an unplanned boron dilution event (Ref. 1) in MODE 6 is precluded by adherence to this LCO which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portions of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required in MODE 6. The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and, thus, avoid a reduction in SHUTDOWN MARGIN.
APPLICABILITY	In MODE 6, this LCO is applicable to prevent an unplanned boron dilution event by ensuring isolation of all sources of unborated water to the RCS. In MODES 1 through 5, the requirements of LCO 3.1.9, “Chemical and Volume Control System (CVS) Demineralized Water Isolation Valves and Makeup Line Isolation Valves,” apply.

BASES

ACTIONS

The ACTIONS Table has been modified by a Note which allows separate Condition entry for each unborated water source flow path.

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the plant in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of “Immediately” shall not preclude completion of actions to establish a safe condition, including movement of a component to a safe location.

Condition A has been modified by a Note to require that Required Action A.3 must be completed whenever Condition A is entered.

A.2

Preventing unplanned dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position verifies that the valves cannot be inadvertently opened. The Completion Time of “Immediately” requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

A.3

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to verify that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

These valves are to be secured closed to isolate possible dilution flow paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water source flow paths are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This surveillance demonstrates that the valves are closed through a system

BASES

SURVEILLANCE REQUIREMENTS (continued)

walkdown. The 31 day Frequency is based on engineering judgement and is considered reasonable in view of other administrative controls that will verify that the valve opening is an unlikely possibility.

REFERENCES

1. Chapter 15, “Accident Analyses.”

2. NUREG-0800, Standard Review Plan, Section 15.4.6, “Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the RCS.”

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND	The source range neutron flux monitors are used to monitor the core reactivity during refueling operations. The source range neutron flux monitors are part of the Protection and Safety Monitoring System (PMS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The 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 six decades of neutron flux ($1 \times 10^{+6}$ cps) with a 5% instrument accuracy. The detectors also provide continuous visual and audible indication in the main control room and an audible alarm in the main control room and containment building.
APPLICABLE SAFETY ANALYSES	Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as those associated with an improperly loaded fuel assembly. During initial fuel loading, or when otherwise required, temporary neutron detectors may be used to provide additional reactivity monitoring (Ref. 2). The potential for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2 (Ref. 1). The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO requires two source range neutron flux monitors to be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.
APPLICABILITY	In MODE 6, the source range neutron flux monitors are required to be OPERABLE to determine possible changes in core reactivity. There are no other direct means available to monitor the core reactivity conditions. In MODES 2, 3, 4, and 5, the source range detectors and associated circuitry are also required to be OPERABLE by LCO 3.3.1, “Reactor Trip System Instrumentation.”

BASES

ACTIONS

A.1 and A.2

Redundancy has been lost if only one source range neutron flux monitor is OPERABLE. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of actions to establish a safe condition.

B.1

If no source range neutron flux monitors are OPERABLE, actions to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, actions shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

If no source range neutron flux monitors are OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are discontinued, the core reactivity condition is stabilized and no changes are permitted until the source range neutron flux monitors are restored to OPERABLE status. This stable condition is confirmed by performing SR 3.9.1.1 to verify that the required boron concentration exists.

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The Frequency of once per 12 hours 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.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is the comparison of the indicated parameter values monitored by each of these instruments. It is based on the assumption that the two indication channels should be consistent for the existing core conditions. Changes in core geometry due to fuel loading can result in significant differences between the source range channels, however each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for these same instruments in LCO 3.3.1, “Reactor Trip System Instrumentation.”

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consisting of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed during the refueling outage.

REFERENCES

1. Chapter 15, "Accident Analysis."
2. Section 14.2.6.1, "Initial Fuel Loading."

B 3.9 REFUELING OPERATIONS

B 3.9.4 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 containment, refueling cavity, refueling canal, fuel transfer canal, and spent fuel pool to retain iodine fission product activity 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 within the values reported in Chapter 15.
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the water level in the refueling cavity and the refueling canal is an initial condition design parameter in the analysis of a fuel-handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 1). The fuel handling accident analysis inside containment is described in Reference 2. This analysis assumes a minimum water level of 23 feet. Refueling Cavity Water Level satisfies Criterion 2 of 10 CFR 50.36(c)(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 the values calculated in Reference 2.
APPLICABILITY	Refueling Cavity Water Level is applicable when moving irradiated fuel assemblies in containment. The LCO minimizes the possibility of radioactive release due to a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not being moved 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 3.7.5, “Spent Fuel Pool Water Level.”

BASES

ACTIONS

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within 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 to safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4 1

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 judgement 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 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

2. Section 15.7.4, “Fuel Handling Accident.”

B 3.9 REFUELING OPERATIONS

B 3.9.5 Containment Penetrations

BASES

BACKGROUND

During movement of irradiated fuel assemblies within containment, potential releases of fission product radioactivity within containment are monitored and filtered or are restricted from escaping to the environment when the LCO requirements are met. Monitoring of potential releases of radiation is performed in accordance with Administrative Controls Section 5.5.2, “Radioactive Effluent Control Program.” In MODES 1, 2, 3, and 4, containment OPERABILITY is addressed in LCO 3.6.1, “Containment.” In MODES 5 and 6, closure capability of containment penetrations is addressed in LCO 3.6.8, “Containment Penetrations.” Since there is no potential for containment pressurization due to a fuel handling accident, the Appendix J leakage criteria and tests are not required in MODES 5 and 6.

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 10 CFR 50.34. For a fuel handling accident, the AP1000 dose analysis does not rely on containment closure to meet the offsite radiation exposure limits. This LCO is provided as an additional level of defense against the possibility of a fission product release from a fuel handling accident.

The containment equipment hatches, which are part of the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. During movement of irradiated fuel assemblies within containment, an equipment hatch is considered closed if the hatch cover is held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

If the equipment hatch is open, an alternative barrier between the containment atmosphere and the outside atmosphere shall be in place. Each containment equipment hatch opens into a staging area in the auxiliary building. These staging areas contain doors that open to the radiologically controlled areas of the annex building. The annex building contains a door that opens to the outside atmosphere. The alternate barrier may consist of the staging area in the auxiliary building, or may consist of the staging areas in the auxiliary building and the radiologically controlled areas in the annex building provided the doors from the annex building to the outside atmosphere are closed. The alternate barrier may

BASES

BACKGROUND (continued)

also consist of a temporary equipment hatch cover that provides equivalent isolation capability. The alternate boundary prevents the airborne fission products from being readily released to the atmosphere if the equipment hatches were open during a fuel handling accident.

If an equipment hatch is open during movement of irradiated fuel assemblies within containment, the containment air filtration system (VFS) shall be OPERABLE, and at least one exhaust fan shall be operating to provide for monitoring of air-borne radioactivity. This system services the containment, and upon detection of high radiation, also services the fuel handling area, the auxiliary building (including the staging areas), and the annex building. If high airborne radioactivity is detected in the area enclosed by the alternate barrier, the radiologically controlled area ventilation system (VAS) supply and exhaust duct isolation dampers automatically close to isolate the affected area from the outside environment, and the VAS exhaust is automatically aligned to the VFS exhaust subsystem. The operation of the VFS exhaust fans provides the system with the ability for monitoring of radioactivity releases from containment following a fuel handling accident and, if operating, will provide filtration of the containment atmosphere.

If a personnel air lock or spare containment penetration is open during movement of irradiated fuel assemblies within containment, then the containment air filtration system (VFS) shall be OPERABLE and operating to monitor for the release of radioactivity and to provide filtration of the air inside containment. These penetrations open into the auxiliary building. Upon detection of high radiation in the exhaust air from the auxiliary building, VFS will provide filtered exhaust of these areas. Considering that these penetrations open into the auxiliary building and not directly to the atmosphere, and that the VFS is in operation, an alternate barrier to the release of radioactivity directly to the environment is provided.

APPLICABLE SAFETY ANALYSES

For the AP1000, there are no safety analyses that require containment closure during movement of irradiated fuel assemblies within containment, other than those discussed in LCO 3.6.8. Fuel handling accidents, analyzed in Reference 1, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.4, “Refueling Cavity Water Level,” ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the

BASES

APPLICABLE SAFETY ANALYSES (continued)

guideline values specified in 10 CFR 50.34. Standard Review Plan, Section 15.0.1 (Reference 2), defines the dose acceptance limit to be 25% of the limiting dose guideline values.

This specification is included as defense-in-depth.

LCO

This LCO provides defense-in-depth against the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. This LCO requires that if an equipment hatch, personnel air lock, or spare containment penetration is open during movement of irradiated fuel assemblies within containment, then the containment air filtration system (VFS) shall be OPERABLE and operating to monitor for the release of radioactivity and to provide filtration of the air inside containment.

The VFS is OPERABLE when:

a. One VFS exhaust fan is operating; the associated HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and air circulation can be maintained;

b. An alternative barrier between the containment atmosphere and the outside atmosphere is in place. The alternate barrier may consist of the staging area in the auxiliary building, or may consist of the staging areas in the auxiliary building and the radiologically controlled areas in the annex building provided the doors from the annex building to the outside atmosphere are closed.

   Doors in the alternate barrier which are normally closed may be opened for short periods of time for ingress and egress. The alternate barrier may also consist of a temporary equipment hatch cover that provides equivalent isolation capability.

APPLICABILITY

The containment penetration requirements are applicable during movement of irradiated fuel assemblies within containment because this is when there is a potential for a 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 irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Containment closure capability in MODES 5 and 6 are addressed by LCO 3.6.8.

ACTIONS

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

The required status for the containment equipment hatch, air locks or spare penetration is either closed, or open with the VFS OPERABLE and operating. The required status for the containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere is either closed by a manual or automatic isolation valve, blind flange or equivalent, or capable of being closed by an OPERABLE Containment Isolation Signal. If the containment equipment hatch or 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 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 3.9.5 1

This Surveillance verifies that each of the containment penetrations required to be in its closed position is in that position or the VFS is OPERABLE and operating. For the VFS to be considered OPERABLE, this surveillance also requires that an alternate barrier is in place.

SR 3.9.5.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. The Frequency is in accordance with the Inservice Testing Program.

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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.5.3

This SR verifies the ability of the VFS to maintain a negative pressure (≤ -0.125 inches water gauge relative to outside atmospheric pressure) in the containment and the portions of the auxiliary and/or annex building that comprise the envelope defined as the alternate barrier. This surveillance is performed with the VFS in containment operating. Doors in the alternate barrier which are normally closed may be opened for ingress and egress. The portion of the VAS which services the area enclosed by the alternate barrier is aligned to the VFS exhaust subsystem, and the VAS auxiliary/annex building supply fans and VFS containment purge supply fans not operating. The Frequency of 24 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 3).

SR 3.9.5.4

The VFS should be checked periodically to ensure that it functions properly. As the operating conditions on this system are not severe, testing each train within 31 days prior to fuel movement provides an adequate check on this system. Operation of the heater dries out any moisture accumulated in the charcoal from humidity in the ambient air.

REFERENCES

1. Section 15.7.4, “Fuel Handling Accident.”
2. NUREG-0800, Section 15.0.1, Rev. 0.
3. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Containment Air Filtration System (VFS)

BASES

BACKGROUND

The radiologically controlled area ventilation system (VAS) serves the fuel handling area of the auxiliary building, and the radiologically controlled portions of the auxiliary and annex buildings, except for the health physics and hot machine shop areas which are provided with a separate ventilation system (VHS). If high airborne radioactivity is detected in the exhaust air from the fuel handling area, the auxiliary building, or the annex buildings, the VAS supply and exhaust duct isolation dampers automatically close to isolate the affected area from the outside environment and the containment air filtration exhaust subsystem starts. The VFS exhaust subsystem prevents exfiltration of unfiltered airborne radioactivity by maintaining the isolated zone at ≤ -0.125 inches water gauge pressure relative to the outside atmosphere. Monitoring of potential releases of radiation is performed in accordance with Administrative Controls Section 5.5.2, “Radioactive Effluent Control Program.”

For a fuel handling accident, the AP1000 dose analysis does not rely on the OPERABILITY of the VAS or VFS exhaust subsystem to meet the offsite radiation exposure limits. This LCO is provided as an additional level of defense-in-depth against the possibility of a fission product release from a fuel handling accident in the fuel building. The plant vent radiation detectors monitor effluents discharged from the plant vent to the environment.

Each VFS exhaust subsystem includes one 100 percent capacity exhaust air filtration unit, and the associated exhaust fan, heater and ductwork.

The filtration units are connected to a ducted system with isolation dampers to provide HEPA filtration and charcoal adsorption of exhaust air from the containment, fuel handling area, radiologically controlled areas of the auxiliary and annex buildings. A gaseous radiation monitor is located downstream of the exhaust air filtration units to provide an alarm if abnormal gaseous releases are detected. The plant vent exhaust flow is monitored for gaseous, particulate and iodine releases to the environment. During conditions of abnormal airborne radioactivity in the fuel handling area, auxiliary and/or annex buildings, the VFS exhaust subsystem provides filtered exhaust to minimize unfiltered offsite releases.

BASES

BACKGROUND (continued)

The VAS is described in Reference 1 and the VFS is described in Reference 2.

APPLICABLE SAFETY ANALYSES

The VFS is not required to mitigate the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident does not assume that the VFS provides a filtered exhaust, and its operation would reduce the consequences of the fuel handling accident.

This specification is included for defense-in-depth.

LCO

One VFS exhaust subsystem is required to be OPERABLE to reduce the consequences of a fuel handling accident by filtering the fuel building atmosphere.

A VFS exhaust subsystem is considered OPERABLE when its associated:

a. Exhaust fan is capable of operating;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function;

c. The associated heater and ductwork are capable of operating.

APPLICABILITY

During movement of irradiated fuel in the fuel handling area, one VFS exhaust subsystem is OPERABLE to alleviate the potential consequences of a fuel handling accident.

ACTIONS

LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. 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 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)

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

When the required VFS exhaust subsystem is inoperable during movement of irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Each VFS exhaust subsystem should be checked 31 days prior to fuel movement in the fuel handling area to ensure that it functions properly. As the operating conditions on this subsystem are not severe, testing each subsystem within one month prior to fuel movement provides an adequate check on this system. Operation of the heater dries out any moisture accumulated in the charcoal from humidity in the ambient air.

SR 3.9.6.2

This SR verifies that the VAS fuel handling area subsystem aligns to the VFS and that the VFS exhaust subsystem starts and operates on an actual or simulated actuation signal. During the post-accident mode of operation, the VAS fuel handling area subsystem aligns to the VFS filtered exhaust subsystem. The 24 month Frequency is consistent with Reference 4.

SR 3.9.6.3

This SR verifies the integrity of the fuel handling area of the auxiliary building enclosure. The ability of the VAS and VFS to maintain negative pressure (≤ -0.125 inches water gauge relative to outside atmospheric pressure) in the fuel handling area of the auxiliary building is periodically tested to verify proper function of the VAS and VFS exhaust subsystem. During this surveillance, the VAS fuel handling area subsystem is aligned

BASES

SURVEILLANCE REQUIREMENTS (continued)

to the operating VFS exhaust subsystem. The fan for the VAS fuel handling area subsystem is off. In this configuration, the VFS exhaust subsystem is designed to maintain a negative pressure in the fuel handling area of the auxiliary building (≤ -0.125 inches water gauge relative to outside atmospheric pressure), to prevent unfiltered and unmonitored leakage. Doors may be opened for short periods of time to allow ingress and egress. During this surveillance, the VAS may be servicing the remaining portions of the auxiliary and annex buildings. The Frequency of 24 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 5).

REFERENCES

1. Section 9.4.3, “Radiologically Controlled Area Ventilation System.”
2. Section 9.4.7, “Containment Air Filtration System.”
3. Section 15.7.4, “Fuel Handling Accident.”
4. Regulatory Guide 1.140 (Rev. 2).
5. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Decay Time

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment or in the fuel handling area inside the auxiliary building requires allowing at least 48 hours for radioactive decay time before fuel assembly handling can be initiated. During fuel handling, this ensures that sufficient radioactive decay has occurred in the event of a fuel handling accident (Refs. 1 and 2). Sufficient radioactive decay of short-lived fission products would have occurred to limit offsite doses from the accident to within the values reported in Chapter 15.
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the radioactivity decay time is an initial condition design parameter in the analysis of a fuel-handling accident inside containment or in the fuel handling area inside the auxiliary building, as postulated by Regulatory Guide 1.183 (Ref. 1). The fuel handling accident analysis inside containment or in the fuel handling area inside the auxiliary building is described in Reference 2. This analysis assumes a minimum radioactive decay time of 48 hours. Radioactive decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	A minimum radioactive decay time of 48 hours is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment or in the fuel handling area inside the auxiliary building are within the values calculated in Reference 2.
APPLICABILITY	Radioactive decay time is applicable when moving irradiated fuel assemblies in containment or in the fuel handling area inside the auxiliary building. The LCO minimizes the possibility of radioactive release due to a fuel handling accident that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not being moved, 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 also covered by LCO 3.7.5, “Spent Fuel Pool Water Level.”

BASES

ACTIONS

LCO 3.0.8 is applicable while in MODE 5 or 6. Since movement of irradiated fuel assemblies with less than 48 hours of decay time can occur in MODE 6 after removing the reactor vessel head following the reactor shutdown, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 6, the fuel movement is independent of shutdown reactor operations since the reactor is already shutdown. Entering LCO 3.0.8 while in MODE 6 would not specify any action.

A.1

With a decay time of less than 48 hours, all operations involving movement of irradiated fuel assemblies within containment or in the fuel handling area inside the auxiliary building shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement to safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification that the reactor has been subcritical for at least 48 hours prior to movement of irradiated fuel in the reactor pressure vessel to the refueling cavity in containment or to the fuel handling area inside the auxiliary building ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Specifying radioactive decay time limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident (Ref. 2).

REFERENCES

1. Regulatory Guide 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

2. Section 15.7.4, “Fuel Handling Accident.”