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 limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO for minimum RCS pressure is consistent with operation within the nominal operating envelope and is above that used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher minimum DNBR. A pressure lower than the minimum specified will cause the plant to approach the DNB limit.

The LCO for maximum RCS coolant hot leg temperature is consistent with full power operation within the nominal operating envelope and is lower than the initial hot leg temperature in the analyses. A hot leg temperature lower than that specified will produce a higher minimum DNBR. A temperature higher than that specified will cause the plant to approach the DNB limit.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. A higher RCS flow rate will produce a higher DNBR. A lower RCS flow will cause the plant to approach the DNB limit.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ [1.3]. This is the acceptance limit for the RCS DNBR parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed for include loss of coolant flow events and dropped or stuck control rod events. A key assumption for the analysis of these events is that the core power distribution is within the

BASES

APPLICABLE SAFETY ANALYSES (continued)

limits of LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE OPERATING LIMITS," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The core outlet pressure assumed in the safety analyses is 2135 psia. The minimum pressure specified in LCO 3.4.1 is the limit value in the reactor coolant loop as measured at the hot leg pressure tap.

The safety analyses are performed with an assumed RCS coolant average temperature of 581°F (579°F plus 2°F allowance for calculational uncertainty). The corresponding hot leg temperature of 604.6°F is calculated by assuming an RCS core outlet pressure of 2135 psia and an RCS flow rate of 374,880 gpm. The maximum temperature specified is the limit value at the hot leg resistance temperature detector.

The safety analyses are performed with an assumed RCS flow rate of 374,880 gpm. The minimum flow rate specified in LCO 3.4.1 is the minimum mass flow rate.

Analyses have been performed to establish the pressure, temperature, and flow rate requirements for three pump and four pump operation. The flow limits for three pump operation are substantially lower than for four pump operation. To meet the DNBR criterion, a corresponding maximum power limit is required (see Bases for LCO 3.4.4, "RCS Loops - MODES 1 and 2").

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables: RCS loop (hot leg) pressure, RCS hot leg temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure and temperature limits are to be applied to the loop with two reactor coolant pumps (RCPs) running for the three RCPs operating condition.

The LCO numerical values for pressure, temperature, and flow rate are given for the measurement location but have not been adjusted for instrument error. Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of this LCO.

BASES

APPLICABILITY

In MODE 1, the limits on RCS pressure, RCS hot leg temperature, and RCS flow rate must be maintained during steady state with four pump or three pump operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES the power level is low enough so that DNB is not a concern.

The Note indicates the limit on RCS pressure may be exceeded during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNBR related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of LCO 3.4.1, but violation of an SL merits a stricter, more severe Required Action. Should a violation of LCO 3.4.1 occur, the operator must check whether an SL may have been exceeded.

ACTIONS

A.1

Loop pressure and hot leg coolant temperature are controllable and measurable parameters. With one or both of these parameters not within the LCO limits, action must be taken to restore the parameters. RCS flow rate is not a controllable parameter and is not expected to vary during steady state four pump or three pump operation. However, if the flow rate is below the LCO limit, the parameter must be restored to within limits or power must be reduced as required in Required Action B.1, to restore DNBR 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, determine the cause for the off normal condition, and restore the readings within limits. The Completion Time is based on plant operating experience.

B.1

If the Required Action A.1 is not met within the 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

BASES

ACTIONS (continued)

6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

The 6 hour Completion Time is reasonable, based on operating experience, to reduce power in an orderly manner in conjunction with even control of steam generator heat removal.

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 for loop (hot leg) pressure is sufficient to ensure that the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The RCS pressure value specified is dependent on the number of pumps in operation and has been adjusted to account for the pressure loss difference between the core exit and the measurement location. The value used in the plant safety analysis is 2135 psia. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation is within safety analysis assumptions.

A Note has been added to indicate the pressure limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

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 hot leg temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

A Note has been added to indicate the temperature limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every [18] months allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow rate.

The Frequency of [18] months reflects the importance of verifying flow after a refueling outage when the core has been altered or RCS flow characteristics may have been modified, which may have caused change of flow.

The Surveillance is modified by a Note that indicates the SR does not need to be performed until stable thermal conditions are established at higher power levels. The Note is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance cannot be performed at low power or in MODE 2 or below because at low power the ΔT across the core will be too small to provide valid results.

REFERENCES

1. FSAR, Chapter [15].

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND	Establishing the value for the minimum temperature for reactor criticality is based upon considerations for: a. Operation within the existing instrumentation ranges and accuracies and b. Operation with reactor vessel above its minimum nil ductility reference temperature when the reactor is critical. The reactor coolant moderator temperature coefficient used in core operating and accident analysis is typically defined for the normal operating temperature range (532°F to 579°F). The Reactor Protection System (RPS) receives inputs from the narrow range hot leg temperature detectors, which have a range of 520°F to 620°F. The integrated control system controls average temperature (T_{avg}) using inputs of the same range. Nominal T_{avg} for making the reactor critical is 532°F. Safety and operating analyses for lower temperatures have not been made.
APPLICABLE SAFETY ANALYSES	There are no accident analyses that dictate the minimum temperature for criticality, but all low power safety analyses assume initial temperatures near the 525°F limit (Ref. 1). The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The purpose of the LCO is to prevent criticality outside the normal operating regime (532°F to 579°F) and to prevent operation in an unanalyzed condition. The LCO limit of 525°F has been selected to be within the instrument indicating range (520°F to 620°F). The limit is also set slightly below the lowest power range operating temperature (532°F).
APPLICABILITY	The reactor has been designed and analyzed to be critical in MODES 1 and 2 only and in accordance with this Specification. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2 when $k_{eff} \geq 1.0$.

BASES

ACTIONS	A.1 With T_{avg} below 525°F, 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 $K_{eff} < 1.0$ in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30 minute period. The Completion Time reflects the ability to perform this Action and maintain the plant within the analyzed range. If T_{avg} can be restored within the 30 minute time period, shutdown is not required.
SURVEILLANCE REQUIREMENTS	SR 3.4.2.1 RCS loop average temperature is required to be verified at or above 525°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. FSAR, Chapter [15].

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, and inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 3).

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 4).

Material toughness properties of the ferritic materials of the reactor vessel are determined in accordance with the NRC Standard Review Plan

BASES

BACKGROUND (continued)

(Ref. 5), ASTM E 185 (Ref. 6), and additional reactor vessel requirements. These properties are then evaluated in accordance with Reference 3.

The actual shift in the nil ductility reference temperature (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. 6) and Appendix H of 10 CFR 50 (Ref. 7). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The calculation to generate the ISLH testing curve uses different safety factors (per Ref. 3) than the heatup and cooldown curves. The ISLH testing curve also extends to the RCS design pressure of 2500 psia.

The P/T limit curves and associated temperature rate of change limits are developed in conjunction with stress analyses for large numbers of operating cycles and provide conservative margins to nonductile failure. Although created to provide limits for these specific normal operations, the curves also can be used to determine if an evaluation is necessary for an abnormal transient.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 8) provides a

BASES

BACKGROUND (continued)

recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Since the P/T limits are not derived from any DBA analysis, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH 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 follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature,

BASES

LCO (continued)

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 Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.2, "RCS Minimum Temperature for Criticality," and Safety Limit (SL) 2.1, "SLs," also provide operational restrictions for pressure and temperature and maximum pressure. MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation to 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 pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. The evaluation must be

BASES

ACTIONS (continued)

completed, documented, and approved in accordance with established plant procedures and administrative controls.

ASME Code, Section XI, Appendix E (Ref. 8) may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline. The evaluation must extend to all components of the RCPB.

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 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a lower MODE because: (a) the RCS remained in an unacceptable pressure and temperature region for an extended period of increased stress, or (b) 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. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours, or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Actions B.1 and B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions. However, if the favorable evaluation is accomplished while

BASES

ACTIONS (continued)

reducing pressure and temperature conditions, a return to power operation may be considered without completing Required Action B.2.

Pressure and temperature are reduced by bringing the plant to 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 MODE from full power conditions in an orderly manner and 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 MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified acceptable 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 within this time in a controlled manner.

In addition to restoring operation to 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 entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analysis, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 8), may also 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 RCPB integrity.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes.

This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that requires this SR to be performed only during system heatup, cooldown, and ISLH testing.

REFERENCES

1. BAW-10046A, Rev. 1, July 1977.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. Regulatory Guide 1.99, Revision 2, May 1988.
5. NUREG-0800, Section 5.3.1, Rev. 1, July 1981.
6. ASTM E 185-82, July 1982.
7. 10 CFR 50, Appendix H.
8. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission,

b. Improving the neutron economy by acting as a reflector,

c. Carrying the soluble neutron poison, boric acid,

d. Providing a second barrier against fission product release to the environment, and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains an SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with either three or four pumps to be in operation. With three pumps in operation the reactor power level is restricted to [79.9]% RTP to preserve the core power to flow relationship, thus maintaining the margin to DNB. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying the minimum number of pumps is an effective technique for designating the proper forced flow rate for heat transport, and specifying two loops provides for the needed amount of heat removal capability for the allowed power levels. Specifying two RCS loops also provides the minimum necessary paths (two SGs) for heat removal.

The Reactor Protection System (RPS) nuclear overpower trip setpoint is automatically reduced when one pump is taken out of service; manual resetting is not necessary.

BASES

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the Design Bases Accident (DBA) 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 pumps in service.

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming either three or four pumps are in operation. The majority of the plant safety analysis is based on initial conditions at high core power or zero power. The accident analyses that are of most importance to RCP operation are the four pump coastdown, single pump locked rotor, and single pump (broken shaft or coastdown) (Ref. 1).

Steady state DNB analysis has been performed for four, three, and two pump combinations. For four pump operation, the steady state DNB analysis, which generates the pressure and temperature SL (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of [112]% RTP. This is the design overpower condition for four pump operation. The [112]% value is the accident analysis setpoint of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The three pump pressure temperature limit is tied to the steady state DNB analysis, which is evaluated each cycle. The flow used is the minimum allowed for three pump operation. The actual RCS flow rate will exceed the assumed flow rate. With three pumps operating, overpower protection is automatically provided by the power to flow ratio of the RPS nuclear overpower based on RCS flow and AXIAL POWER IMBALANCE setpoint. The maximum power level for three pump operation is [79.9]% RTP and is based on the three pump flow as a fraction of the four pump flow at full power.

Although the Specification limits operation to a minimum of three pumps total, existing design analyses show that operation with one pump in each loop (two pumps total) is acceptable when core THERMAL POWER is restricted to be proportionate to the flow. However, continued power operation with two RCPs removed from service is not allowed by this Specification.

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by the number of RCPs in operation in both RCS loops for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power; if only three pumps are available, power must be reduced.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and has the potential to produce maximum THERMAL POWER. To ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5,	"RCS Loops - MODE 3,"
LCO 3.4.6,	"RCS Loops - MODE 4,"
LCO 3.4.7,	"RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8,	"RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4,	"Decay Heat Removal (DHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.5,	"Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.4.4.1</u> This SR requires verification every 12 hours of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.
REFERENCES	1. FSAR, Chapter [ ].

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the steam generators (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.

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation for heat removal during heatup and cooldown. The number of RCPs in operation will vary depending on operational needs, and the intent of this LCO is to provide forced flow from at least one RCP for core heat removal and transport. The flow provided by one RCP is adequate for heat removal and for boron mixing. However, two RCS loops are required to be OPERABLE to provide redundant paths for heat removal.

Reactor coolant natural circulation is not normally used; however, the natural circulation flow rate is sufficient for core cooling. If entry into natural circulation is required, the reactor coolant at the highest elevation of the hot leg must be maintained subcooled for single phase circulation. When in natural circulation, it is preferable to remove heat using both SGs to avoid idle loop stagnation that might occur if only one SG were in service. One generator will provide adequate heat removal. Boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 3.

Failure to provide heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops - MODE 3 satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require two loops to be available for heat removal thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although

BASES

LCO (continued)

natural circulation flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation.

The Note permits a limited period of operation without RCPs. All RCPs may not be in operation for ≤ 8 hours per 24 hour period for the transition to or from the Decay Heat Removal (DHR) System, and otherwise may be de-energized for ≤ 1 hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, boron reduction with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least [10]°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation (e.g., change operation from one DHR train to the other, to perform surveillance or startup testing, to perform the transition to and from DHR System cooling, or to avoid operation below the RCP minimum net positive suction head limit). The time period is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

Operation in other MODES is covered by:

LCO 3.4.4,	"RCS Loops - MODES 1 and 2,"
LCO 3.4.6,	"RCS Loops - MODE 4,"
LCO 3.4.7,	"RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8,	"RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4,	"Decay Heat Removal (DHR) and Coolant Circulation - High Water Level" (MODE 6), and

BASES

APPLICABILITY (continued)

LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

If one RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the RCS loop to OPERABLE status within a Completion Time of 72 hours. This time allowance is a justified period to be without the redundant nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core.

B.1

If restoration of an RCS loop as required in A.1 is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the plant may be placed on the DHR System. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to achieve cooldown and depressurization from the existing plant conditions and without challenging plant systems.

C.1 and C.2

If two RCS loops are inoperable or a required RCS loop is not in operation, except as provided in the Note in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be immediately suspended. Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to OPERABLE status and to operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required number of loops and pumps is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.5.2

Verification that each required RCP is OPERABLE ensures that the single failure criterion is met and that an additional RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the steam generators (SGs) or decay heat removal (DHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either reactor coolant pumps (RCPs) or DHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RCP or one DHR pump for decay heat removal and transport. The flow provided by one RCP or one DHR pump is adequate for heat removal. The other intent of this LCO is to require that two paths (loops) be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial condition in MODE 4.

RCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that two loops, RCS or DHR, be OPERABLE in MODE 4 and one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS or DHR System loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. The second loop that is required to be OPERABLE provides redundant paths for heat removal.

The Note permits a limited period of operation without RCPs. All RCPs may not be in operation for ≤ 8 hours per 24 hour period for the transition to or from the DHR System and otherwise may be de-energized for ≤ 1 hour per 8 hour period. This means that natural circulation has been established using the SGs. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

BASES

LCO (continued)

The Note also permits the DHR pumps to be stopped for ≤ 1 hour per 8 hour period. When the DHR pumps are stopped, no alternate heat removal path exists, unless the RCS and SGs have been placed in service in forced or natural circulation. The response of the RCS without the DHR System depends on the core decay heat load and the length of time that the DHR pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by DHR, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) or low temperature overpressure protection (LTOP) limits) must be observed and forced DHR flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both DHR trains are to be limited to situations where:

a. Pressure and pressure and temperature increases can be maintained well within the allowable pressure (P/T and LTOP) and 10°F subcooling limits or

b. An alternate heat removal path through the SG is in operation.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

Similarly for the DHR System, an OPERABLE DHR loop is comprised of the OPERABLE DHR pump(s) capable of providing forced flow to the DHR heat exchanger(s). DHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs or the DHR System.

Operation in other MODES is covered by:

LCO 3.4.4,	"RCS Loops - MODES 1 and 2,"
LCO 3.4.5,	"RCS Loops - MODE 3,"
LCO 3.4.7,	"RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8,	"RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4,	"Decay Heat Removal (DHR) and Coolant Circulation - High Water Level" (MODE 6), and

BASES

APPLICABILITY (continued)

LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

If only one required RCS loop or DHR loop is OPERABLE and in operation, redundancy for heat removal is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

A.2

If restoration is not accomplished and a DHR loop is OPERABLE, the unit must be brought to MODE 5 within the following 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one DHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining DHR loop, it would be safer to initiate that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems.

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if a DHR loop is OPERABLE. With no DHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on the restoration of a DHR loop, rather than a cooldown of extended duration.

B.1 and B.2

If two required RCS or DHR loops are inoperable or a required loop is not in operation, except during conditions permitted by the Note in the LCO section, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RCS or DHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must continue until one loop is restored to operation.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance requires verification every 12 hours of the required DHR or RCS loop in operation to ensure forced flow is providing decay heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.6.2

Verification that each required pump is OPERABLE ensures that an additional RCS or DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant or the component cooling water via the decay heat removal (DHR) heat exchangers. While the principal means for decay heat removal is via the DHR System, the SGs are specified as a backup means for redundancy. Although the SGs cannot remove heat unless steaming occurs (which is not possible in MODE 5), they are available as a temporary heat sink and can be used by allowing the RCS to heat up into the temperature region of MODE 4 where steaming can be effective for heat removal. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, DHR loops are the principal means for heat removal. The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one DHR loop for decay heat removal and transport. The flow provided by one DHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for either SG heat removal or DHR System heat removal. In this MODE, reactor coolant pump (RCP) operation may be restricted because of net positive suction head (NPSH) limitations, and the SG will not be able to provide steam for the turbine driven feed pumps. However, to ensure that the SGs can be used as a heat sink, a motor driven feedwater pump is needed, because it is independent of steam. Condensate pumps, startup pumps, or the motor driven auxiliary feedwater pump can be used. If RCPs are available, the steam generator level need not be adjusted. If RCPs are not available, the water level must be adjusted for natural circulation. The high entry point in the generator should be accessible from the feedwater pumps so that natural circulation can be stimulated. The SGs are primarily a backup to the DHR pumps, which are used for forced flow. By requiring the SGs to be a backup heat removal path, the option to increase RCS pressure and temperature for heat removal in MODE 4 is provided.

BASES

APPLICABLE SAFETY ANALYSES	No safety analyses are performed with initial conditions in MODE 5. RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least one of the DHR loops be OPERABLE and in operation with an additional DHR loop OPERABLE or both SGs with secondary side water level ≥ [50]%. One DHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second DHR loop is normally maintained as a backup to the operating DHR loop to provide redundancy for decay heat removal. However, if the standby DHR loop is not OPERABLE, a sufficient alternate method of providing redundant heat removal paths is to provide both SGs with their secondary side water levels ≥ [50]%. Should the operating DHR loop fail, the SGs could be used to remove the decay heat.

Note 1 permits the DHR pumps to not be in operation for up to 1 hour per 8 hour period. The circumstances for stopping both DHR trains are to be limited to situations where: (a) Pressure and temperature increases can be maintained well within the allowable pressure (P/T and low temperature overpressure protection) and 10°F subcooling limits or (b) Alternate heat paths through the SGs are in operation.

The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when DHR forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the generators are used as a backup for decay heat removal and, to ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

In MODE 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation. This is permitted to change operation from one DHR train to the other, perform surveillance or startup testing, perform the transition to and from the DHR System, or to avoid operation below the RCP minimum NPSH limit. The time period is acceptable because natural circulation is acceptable for heat removal, the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

BASES

LCO (continued)

Note 2 allows one DHR loop to be inoperable for a period of up to 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting DHR loops to not be in operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the DHR loops.

An OPERABLE DHR loop is composed of an OPERABLE DHR pump and an OPERABLE DHR heat exchanger.

DHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with loops filled, forced circulation is provided by this LCO to remove decay heat from the core and to provide proper boron mixing. One loop of DHR provides sufficient circulation for these purposes.

Operation in other MODES is covered by:

LCO 3.4.4,	"RCS Loops - MODES 1 and 2,"
LCO 3.4.5,	"RCS Loops - MODE 3,"
LCO 3.4.6,	"RCS Loops - MODE 4,"
LCO 3.4.8,	"RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4,	"Decay Heat Removal (DHR) and Coolant Circulation - High Water Level" (MODE 6), and
LCO 3.9.5,	"Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1, A.2, B.1, and B.2

If one DHR loop is OPERABLE and any required SG has secondary side water level < [50]%, redundancy for heat removal is lost. Action must be initiated to restore the inoperable (non-operating) DHR loop to OPERABLE status or initiate action to restore the secondary side water level in the SGs, and action must be taken immediately. Either Required Action will restore redundant decay heat removal paths. The immediate

BASES

ACTIONS (continued)

Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

C.1 and C.2

If no required DHR loop is in operation, except as provided in Note 1, or no required DHR loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore a DHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required DHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ [50]% ensures that redundant heat removal paths are available if the second DHR loop is not OPERABLE. If both DHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.7.3

Verification that each required DHR pump is OPERABLE ensures that redundant paths for heat removal are available. The requirement also ensures that the additional loop can be placed in operation if needed to

BASES

SURVEILLANCE REQUIREMENTS (continued)

maintain decay heat removal and reactor coolant circulation. If the secondary side water level is ≥ [50]% in both SGs, this Surveillance is not needed. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the decay heat removal (DHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

Loops are not filled when the reactor coolant water level is within the horizontal portion of the hot leg as might be the case for refueling or maintenance on the reactor coolant pumps or SGs. GL 88-17 (Ref. 1) expresses concerns for loss of decay heat removal for this operating condition. With water at this low level, the margin above the decay heat suction piping connection to the hot leg is small. The possibility of loss of level or inlet vortexing exists and if it were to occur, the operating DHR pump could become air bound and fail resulting in a loss of forced flow for heat removal. As a consequence the water in the core will heat up and could boil with the possibility of core uncovering due to boil off. Because the containment hatch may be open at this time, a pathway to the outside for fission product release exists if core damage were to occur.

In MODE 5 with loops not filled, only DHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one DHR pump for decay heat removal and transport, to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 5 with loops not filled. The flow provided by one DHR pump is adequate for heat removal and for boron mixing.

RCS Loops - MODE 5 (Loops Not Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that a minimum of two DHR loops be OPERABLE and that one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the DHR system unless forced flow is used. A minimum of one running

BASES

LCO (continued)

decay heat removal pump meets the LCO requirement for one loop in operation. An additional DHR loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits the DHR pumps to not be in operation for ≤ 15 minutes when switching from one train to the other. The circumstances for stopping both DHR pumps are to be limited to situations where the outage time is short [and temperature is maintained ≤ [160]°F]. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained or draining operations when DHR forced flow is stopped.

Note 2 allows one DHR loop to be inoperable for a period of 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE DHR loop is composed of an OPERABLE DHR pump capable of providing forced flow to an OPERABLE DHR heat exchanger. DHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the DHR System.

Operation in other MODES is covered by:

LCO 3.4.4,	"RCS Loops - MODES 1 and 2,"
LCO 3.4.5,	"RCS Loops - MODE 3,"
LCO 3.4.6,	"RCS Loops - MODE 4,"
LCO 3.4.7,	"RCS Loops - MODE 5, Loops Filled,"
LCO 3.9.4,	"Decay Heat Removal (DHR) and Coolant Circulation - High Water Level" (MODE 6), and
LCO 3.9.5,	"Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

If one required DHR loop is inoperable, redundancy for heat removal is lost. Required Action A.1 is to immediately initiate activities to restore a second loop to OPERABLE status. The immediate Completion Time

BASES

ACTIONS (continued)

reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required loop is OPERABLE or the required loop is not in operation, except as provided by Note 1 in the LCO, the Required Action requires immediate suspension of all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 and requires initiation of action to immediately restore one DHR loop to OPERABLE status and operation. The Required Action for restoration does not apply to the condition of both loops not in operation when the exception Note in the LCO is in force. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operations for decay heat removal. The action to restore must continue until one loop is restored.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification every 12 hours that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that redundancy for heat removal is provided. The requirement also ensures that an additional loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other

BASES

SURVEILLANCE REQUIREMENTS (continued)

administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. Generic Letter 88-17, October 17, 1988.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls and emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves (PORVs) are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valve (PORV)," respectively.

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control during normal operation and proper pressure response for anticipated design basis transients. The water level limit thus serves two purposes:

a. Pressure control during normal operation maintains subcooled reactor coolant in the loops and thus is in the preferred state for heat transport and

b. By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes that could result in degraded ability for pressure control.

The maximum water level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus ensuring that pressure relief devices (PORVs or code safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design Safety Limit (SL) of 2750 psig or damage may occur to the PORVs or pressurizer code safety valves.

BASES

BACKGROUND (continued)

The pressurizer heaters are used to maintain a pressure in the RCS so reactor coolant in the loops is subcooled and thus in the preferred state for heat transport to the steam generators (SGs). This function must be maintained with a loss of offsite power. Consequently, the emphasis of this LCO is to ensure that the essential power supplies and the associated heaters are adequate to maintain pressure for RCS loop subcooling with an extended loss of offsite power.

A minimum required available capacity of [126] kW ensures that the RCS pressure can be maintained. Unless adequate heater capacity is available, reactor coolant subcooling cannot be maintained indefinitely. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to loss of single phase natural circulation and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1 and 2, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No safety analyses are performed in lower MODES. 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 the FSAR 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.

The maximum level limit is of prime interest for the loss of main feedwater (LOMFW) event. Conservative safety analyses assumptions for this event indicate that it produces the largest increase of pressurizer level caused by a moderate frequency event. Thus this event has been selected to establish the pressurizer water level limit. Assuming proper response action by emergency systems, the level limit prevents water relief through the pressurizer safety valves. Since prevention of water relief is a goal for abnormal transient operation, rather than an SL, the value for pressurizer level is nominal and is not adjusted for instrument error.

Evaluations performed for the design basis large break loss of coolant accident (LOCA), which assumed a higher maximum level than assumed for the LOMFW event, have been made. The higher pressurizer level assumed for the LOCA is the basis for the volume of reactor coolant released to the containment. The containment analysis performed using

BASES

APPLICABLE SAFETY ANALYSES (continued)

the mass and energy release demonstrated that the maximum resulting containment pressure was within design limits.

The requirement for emergency power supplies is based on NUREG-0737 (Ref. 1). The intent is to allow maintaining the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power. While loss of offsite power is an initial condition or coincident event assumed in many accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated as part of FSAR accident analyses.

The maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 1), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water level ≤ [290] inches ensures that a steam bubble exists. Limiting the 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.

The LCO requires a minimum of [126] kW of pressurizer heaters OPERABLE [and capable of being powered from an emergency power supply]. As such, the LCO addresses both the heaters and the power supplies. The minimum heater capacity required is sufficient to maintain the system near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of [126] kW is derived from the use of nine heaters rated at 14 kW each. The amount needed to maintain pressure is dependent on the insulation losses, which can vary due to tightness of fit and condition.

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

BASES

APPLICABILITY (continued)

has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3 and, for pressurizer water level, for MODE 4 with RCS temperature ≥ [275]°F. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbations, such as reactor coolant pump startup. The temperature of [275]°F has been designated as the cutoff for applicability because LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," provides a requirement for pressurizer level below [275]°F. The LCO does not apply to MODE 5 with loops filled because LCO 3.4.12 applies. The LCO does not apply to MODES 5 and 6 with partial loop operation.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. The Applicability is modified by a Note stating that the OPERABILITY requirements on pressurizer heaters do not apply in MODE 4. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Decay Heat Removal System is in service, and therefore the LCO is not applicable.

ACTIONS

A.1

With pressurizer water level in excess of the maximum limit, action must be taken to restore pressurizer operation to within the bounds assumed in the analysis. This is done by restoring the pressurizer water level to within the limit. The 1 hour Completion Time is considered to be a reasonable time for draining excess liquid.

B.1 and B.2

If the water level cannot be restored, reducing core power constrains heat input effects that drive pressurizer insurge that could result from an anticipated transient. By shutting down the reactor and reducing reactor coolant temperature to at least MODE 3, the potential thermal energy of the reactor coolant mass for LOCA mass and energy releases is reduced.

Six hours is a reasonable time based upon operating experience to reach MODE 3 from full power without challenging plant systems and operators. Further pressure and temperature reduction to MODE 4 with RCS

BASES

ACTIONS (continued)

temperature ≤ [275]°F places the plant into a MODE where the LCO is not applicable. The [24] hour Completion Time to reach the nonapplicable MODE is reasonable based upon operating experience.

C.1

If the [emergency] power supplies to the heaters are not capable of providing [126] kW, or the pressurizer heaters are inoperable, restoration is required in 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power will not occur in this period. Pressure control may be maintained during this time using normal station powered heaters.

D.1 and D.2

If pressurizer heater capability cannot be restored within the allowed Completion Time of Required Action C.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within the following 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable based on operating experience to achieve power reduction from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer water level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess 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.

[SR 3.4.9.2

The SR requires the power supplies are capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Frequency of [[18] months] is considered adequate

BASES

SURVEILLANCE REQUIREMENTS (continued)

to detect heater degradation and has been shown by operating experience to be acceptable.]

[SR 3.4.9.3

This SR is not applicable if the heaters are permanently powered by 1E power supplies.

This Surveillance demonstrates that the heaters can be manually transferred to, and energized by, emergency power supplies. The Frequency of [18] months is based on a typical fuel cycle and is consistent with similar verifications of emergency power.]

REFERENCES 1. NUREG-0737, November 1980.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. Two safety valves are used for MODE 3 and portions of MODE 4. For the remainder of MODE 4, MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. 1). The required lift pressure is 2500 psig ± 1%. The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The upper and lower pressure limits are based on the ± 1% tolerance requirement for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The 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 pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident analyses in the FSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis (Ref. 1) is also based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 1%). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod,

BASES

APPLICABLE SAFETY ANALYSES (continued)

loss of main feedwater, or main feedwater line break accident. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

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 psig) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code 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 both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP cut in temperature, 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 both safety valves for protection.

The LCO is not applicable in MODE 4 when any RCS cold leg temperature is ≤ [283]°F and MODE 5 because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.

The Note allows entry into MODES 3 and 4 with the lift settings 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

BASES

APPLICABILITY (continued)

condition. Only one valve at a time will be removed from service for testing. The [36] hour exception is based on an 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 timeframe.

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 RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperature ≤ [283]°F within 12 hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Similarly, the [24] hours allowed is reasonable, based on operating experience, to reach MODE 4 without challenging plant systems. With any RCS cold leg temperature at or below [283]°F, overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 1), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± [3]% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

BASES

REFERENCES	1. ASME, Boiler and Pressure Vessel Code, Section III, Section XI.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valve (PORV)

BASES

BACKGROUND

The pressurizer is equipped with three devices for pressure relief functions: two American Society of Mechanical Engineers (ASME) pressurizer safety valves that are safety grade components and one PORV that is not a safety grade device. The PORV is an electromatic pilot operated valve that is automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. The PORV may also be manually operated using controls installed in the control room.

An electric motor operated, normally open, block valve is installed between the pressurizer and the PORV. The function of the block valve is to isolate the PORV. Block valve closure is accomplished manually using controls in the control room and may be used to isolate a leaking PORV to permit continued power operation. Most importantly, the block valve is to be used to isolate a stuck open PORV to isolate the resulting small break loss of coolant accident (LOCA). Closure terminates the RCS depressurization and coolant inventory loss.

The PORV, its block valve, and their controls are powered from normal power supplies but are also capable of being powered from emergency supplies. Power supplies for the PORV are separate from those for the block valve. Power supply requirements are defined in NUREG-0737, Paragraph III, G.1 (Ref. 1).

The PORV setpoint is above the high pressure reactor trip setpoint and below the opening setpoint for the pressurizer safety valve as required by IE Bulletin 79-05B (Ref. 2). The purpose of the relationship of these setpoints is to limit the number of transient pressure increase challenges that might open the PORV, which, if opened, could fail in the open position. A pressure increase transient would cause a reactor trip, reducing core energy, and for many expected transients, prevent the pressure increase from reaching the PORV setpoint. The PORV setpoint thus limits the frequency of challenges from transients and limits the possibility of a small break LOCA from a failed open PORV.

Placing the setpoint below the pressurizer safety valve opening setpoint reduces the frequency of challenges to the safety valves, which, unlike the PORV, cannot be isolated if they were to fail open. The PORV setpoint is therefore important for limiting the possibility of a small break LOCA.

BASES

BACKGROUND (continued)

The primary purpose of this LCO is to ensure that the PORV and the block valve are operating correctly so the potential for a small break LOCA through the PORV pathway is minimized, or if a small break LOCA were to occur through a failed open PORV, the block valve could be manually operated to isolate the path.

The PORV may be manually operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available; a condition that would be encountered during loss of offsite power. Steam generator tube rupture (SGTR) is one event that may require use of the PORV if the sprays are unavailable.

The PORV may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORV functions as an automatic overpressure device and limits challenges to the safety valves. Although the PORV acts as an overpressure device for operational purposes, safety analyses [do not take credit for PORV actuation, but] do take credit for the safety valves.

The PORV also provides low temperature overpressure protection (LTOP) during heatup and cooldown. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," addresses this function.

APPLICABLE SAFETY ANALYSES

The PORV small break LOCA break size is bounded by the spectrum of piping breaks analyzed for plant licensing. Because the PORV small break LOCA is located at the top of the pressurizer, the RCS response characteristics are different from RCS loop piping breaks; analyses have been performed to investigate these characteristics.

The possibility of a small break LOCA through the PORV is reduced when the PORV flow path is OPERABLE and the PORV opening setpoint is established to be reasonably remote from expected transient challenges. The possibility is minimized if the flow path is isolated.

The PORV opening setpoint has been established in accordance with Reference 2. It has been set so expected RCS pressure increases from anticipated transients will not challenge the PORV, minimizing the possibility of a small break LOCA through the PORV.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Overpressure protection is provided by safety valves, and analyses do not take credit for the PORV opening for accident mitigation.

Operational analyses that support the emergency operating procedures utilize the PORV to depressurize the RCS for mitigation of SGTR when the pressurizer spray system is unavailable (loss of offsite power). FSAR safety analyses for SGTR have been performed assuming that offsite power is available and thus pressurizer sprays (or the PORV) are available.

The PORV and its block valve satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires the PORV and its associated block valve to be OPERABLE. The block valve is required to be OPERABLE so it may be used to isolate the flow path if the PORV is not OPERABLE. If the block valve is not OPERABLE, the PORV may be used for temporary isolation.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. A likely cause for PORV LOCA is a result of pressure increase transients that cause the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the steam generators are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODES 1 and 2.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the applicability is pertinent to MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

BASES

ACTIONS

A.1 and A.2

With the PORV inoperable, the PORV must be restored or the flow path isolated within 1 hour. The block valve should be closed and power must be removed from the block valve to reduce the potential for inadvertent PORV opening and depressurization.

B.1 and B.2

If the block valve is inoperable, it must be restored to OPERABLE status within 1 hour. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to close the block valve and remove power within 1 hour rendering the PORV isolated. The 1 hour Completion Times are consistent with an allowance of some time for correcting minor problems, restoring the valve to operation, and establishing correct valve positions and restricting the time without adequate protection against RCS depressurization.

C.1 and C.2

If the Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Similarly, the 12 hours allowed is reasonable, based on operating experience, to reach MODE 4 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that it can be closed if needed. The basis for the Frequency of 92 days is ASME Code, Section XI (Ref. 3). Block valve cycling, as stated in the Note, is not required to be performed when it is closed for isolation; cycling could increase the hazard of an existing degraded flow path.

SR 3.4.11.2

PORV cycling demonstrates its function. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.3

This Surveillance is not required for plants with permanent 1E power supplies to the valves.

This SR demonstrates that emergency power can be provided and is performed by transferring power from the normal supply to the emergency supply and cycling the valves. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. NUREG-0737, Paragraph III, G.1, November 1980.
2. NRC IE Bulletin 79-05B, April 21, 1979.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

- REVIEWER'S NOTE -

For plants for which the NRC has approved LTOP setpoints based on non-10 CFR 50, Appendix G, methodology, as allowed in NRC Generic Letter 88-11, the following Bases must be revised accordingly.

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) requirements of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for providing such protection. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 limits.

The reactor vessel material is less tough at reduced temperatures than at normal operating temperature. Also, as vessel neutron irradiation accumulates, the material becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure must be maintained low when temperature is low and must be increased only as temperature is increased.

Operational maneuvering during cooldown, heatup, or any anticipated operational occurrence must be controlled to not violate LCO 3.4.3. Exceeding these limits could lead to brittle fracture of the reactor vessel. LCO 3.4.3 presents requirements for administrative control of RCS pressure and temperature to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection in the applicable MODES by ensuring an adequate pressure relief capacity and a minimum coolant addition capability. The pressure relief capacity requires either the power operated relief valve (PORV) lift setpoint to be reduced and pressurizer coolant level at or below a maximum limit or the RCS depressurized and with an RCS vent of sufficient size to handle the limiting transient during LTOP.

The LTOP approach to protecting the vessel by limiting coolant addition capability allows a maximum of [one] makeup pump, and requires deactivating HPI, and isolating the core flood tanks (CFTs).

BASES

BACKGROUND (continued)

Should more than [one] HPI pump inject on an HPI actuation, the pressurizer level and PORV or another RCS vent cannot prevent overpressurizing the RCS. Even with only one HPI pump OPERABLE, the vent cannot prevent RCS overpressurization.

The pressurizer level limit provides a compressible vapor space or cushion (either steam or nitrogen) that can accommodate a coolant insurge and prevent a rapid pressure increase, allowing the operator time to stop the increase. The PORV, with reduced lift setting, or the RCS vent is the overpressure protection device that acts as backup to the operator in terminating an increasing pressure event.

With HPI deactivated, the ability to provide RCS coolant addition is restricted. To balance the possible need for coolant addition, the LCO does not require the Makeup System to be deactivated. Due to the lower pressures associated with the LTOP MODES and the expected decay heat levels, the Makeup System can provide flow with the OPERABLE makeup pump through the makeup control valve.

<u>PORV Requirements</u>

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit set in the LTOP actuation circuit. The LTOP actuation circuit monitors RCS pressure and determines when an overpressure condition is approached. When the monitored pressure meets or exceeds the setting, the PORV is signaled to open. Maintaining the setpoint within the limits of the LCO ensures the Reference 1 limits will be met in any event analyzed for LTOP.

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

<u>RCS Vent Requirements</u>

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at ambient containment pressure in an RCS overpressure transient, if the relieving requirements of the maximum credible LTOP transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow of the limiting LTOP

BASES

BACKGROUND (continued)

transient and maintaining pressure below 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, it requires removing a pressurizer safety valve, locking the PORV in the open position and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent valve. 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. 3) demonstrate that the reactor vessel can be adequately protected against overpressurization transients during shutdown. In MODES 1, 2, and 3, and in MODE 4 with RCS temperature exceeding [283]°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At nominally [283]°F and below, overpressure prevention falls to an OPERABLE PORV and a restricted coolant level in the pressurizer or to a depressurized RCS and a sufficient size RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP System will be re-evaluated to ensure that its functional requirements can still be met with the PORV and pressurizer level method or the depressurized and vented RCS condition.

Transients that are capable of overpressurizing the RCS have been identified and evaluated. These transients relate to either mass input or heat input: actuating the HPI System, discharging the CFTs, energizing the pressurizer heaters, failing the makeup control valve open, losing decay heat removal, starting a reactor coolant pump (RCP) with a large temperature mismatch between the primary and secondary coolant systems, and adding nitrogen to the pressurizer.

HPI actuation and CFT discharge are the transients that result in exceeding P/T limits within < 10 minutes, in which time no operator action is assumed to take place. In the rest, operator action after that time precludes overpressurization. The analyses demonstrate that the time allowed for operator action is adequate, or the events are self limiting and do not exceed P/T limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The following are required during the LTOP MODES to ensure that transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Deactivating all but [one] makeup pump,

b. Deactivating HPI, and

c. Immobilizing CFT discharge isolation valves in their closed positions.

The Reference 3 analyses demonstrate the PORV can maintain RCS pressure below limits when only one makeup pump is actuated. Consequently, the LCO allows only [one] makeup pump to be OPERABLE in the LTOP MODES.

Since the PORV cannot do this for one HPI pump and the RCS vent cannot do this for even one pump, the LCO also requires the HPI actuation circuits deactivated and the CFTs isolated.

The isolated CFTs must have their discharge valves closed and the valve power breakers fixed in their open positions. The analyses show the effect of CFT discharge is over a narrower RCS temperature range (175°F and below) than that of the LCO ([283]°F and below).

Fracture mechanics analyses established the temperature of LTOP Applicability at [283]°F. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 21 effective full power years (EFPYs) of operation.

This LCO will deactivate the HPI actuation when the RCS temperature is ≤ [283]°F. The consequences of a small break LOCA in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 4 and 5), requirements by having a maximum of [one] makeup pump OPERABLE.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORV is set to open at ≤ [555] psig. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient of uncontrolled HPI actuation of one pump. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoint at or below the derived limit ensures the Reference 1 limits will be met.

The PORV setpoint will be re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement induced by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations.

The PORV is considered an active component. Therefore, its failure represents the worst case LTOP single active failure.

Pressurizer Level Performance

Analyses of operator response time show that the pressurizer level must be maintained ≤ [220] inches to provide the 10 minute action time for correcting transients.

The pressurizer level limit will also be re-evaluated for compliance each time P/T limit curves are revised based on the results of the vessel material surveillance.

RCS Vent Performance

With the RCS depressurized, analyses show a vent of [0.75] square inches is capable of mitigating the transient resulting from full opening of the makeup control valve while the makeup pump is providing RCS makeup. The capacity of a vent this size is greater than the flow resulting from this credible transient at 100 psig back pressure, which is less than the maximum RCS pressure on the P/T limit curve in LCO 3.4.3.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS vent size will also be re-evaluated for compliance each time P/T limit curves are revised based on the results of the vessel material surveillance.

The 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

The LCO requires an LTOP System OPERABLE with a limited coolant input capability and a pressure relief capability. To limit coolant input, the LCO requires a maximum of [one] makeup pump OPERABLE, the HPI deactivated, and the CFT discharge isolation valves closed and immobilized. For pressure relief, it requires either the pressurizer coolant at or below a maximum level and the PORV OPERABLE with a lift setting at the LTOP limit or the RCS depressurized and a vent established.

The LCO is modified by two Notes. Note 1 allows [two makeup pumps] to be made capable of injecting for ≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and surveillance requirements associated with the swap. The intent is to minimize the actual time that more than [one] charging pump is physically capable of injection. Note 2 states that CFT isolation is only required when the CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This Note permits the CFT discharge valve surveillance performed only under these pressure and temperature conditions.

The pressurizer is OPERABLE with a coolant level ≤ [220] inches.

The PORV is OPERABLE when its block valve is open, its lift setpoint is set at ≤ [555] psig and testing has proven its ability to open at that setpoint, and motive power is available to the two valves and their control circuits.

For the depressurized RCS, an RCS vent is OPERABLE when open with an area of at least [0.75] square inches.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ [283]°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The Applicability temperature of [283]°F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure

BASES

APPLICABILITY (continued)

protection to meet LCO 3.4.3 P/T limits above [283]°F. With the vessel head off, overpressurization is not possible.

LCO 3.4.3 provides the operational P/T limits for all MODES.
LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above [283]°F.

ACTIONS

A.1 and B.1

With two or more makeup pumps capable of injecting into the RCS or if the HPI is activated, immediate actions are required to render the other pump(s) inoperable or to deactivate HPI. Emphasis is on immediate deactivation because inadvertent injection with [one] or more HPI pump OPERABLE is the event of greatest significance, since it causes the greatest pressure increase in the shortest time. Also, the vent cannot mitigate overpressurization from the injection of even one HPI pump.

The immediate Completion Times reflect the urgency of quickly proceeding with the Required Actions.

C.1, D.1, and D.2

An unisolated CFT requires isolation within 1 hour only when the CFT pressure is at or more than the maximum RCS pressure for the existing temperature allowed in LCO 3.4.3.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in 12 hours. By increasing the RCS temperature to > 175°F, the CFT pressure of 600 psig cannot exceed the LTOP limits if both tanks are fully injected. Depressurizing the CFTs below the LTOP limit of [555] psig also prevents exceeding the LTOP limits in the same event.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that a limiting LTOP event is not likely in the allowed times.

BASES

ACTIONS (continued)

E.1, F.1, and F.2

With the pressurizer level more than [220] inches, the time for operator action in a pressure increasing event is reduced. The postulated event most affected in the LTOP MODES is failure of the makeup control valve, which fills the pressurizer relatively rapidly. Restoration is required within 1 hour.

If restoration within 1 hour in either case cannot be accomplished, Required Actions F.1 and F.2 must be performed within 12 hours to close the makeup control valve and its isolation valve. These Required Actions limit the makeup capability, which is not required with a high pressurizer level, and permit cooldown and depressurization to continue. Heatup must be stopped because heat addition decreases the reactor coolant density and increases the pressurizer level.

The Completion Times again are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that a limiting LTOP transient is not likely in the allowed times.

G.1, H.1, and H.2

With the PORV inoperable, overpressure relieving capability is lost, and restoration of the PORV within 1 hour is required. If that cannot be accomplished, the ability of the Makeup System to add water must be limited within the next 12 hours.

If restoration cannot be completed within 1 hour, Required Action H.1 and Required Action H.2 must be performed to limit RCS water addition capability. Makeup is not deactivated to maintain the RCS coolant level. Required Action H.1 and Required Action H.2 require reducing the makeup tank level to 70 inches and deactivating the low low makeup tank level interlock to the borated water storage tank. This makes the available makeup water volume insufficient to exceed the LTOP limit by a makeup control valve full opening.

These Completion Times also consider these activities can be accomplished in these time periods. A limiting LTOP event is not likely in those times.

BASES

ACTIONS (continued)

Some PORV testing or maintenance can only be performed at plant shutdown. Such activity is permitted if Required Action H.1 and Required Action H.2 are taken to compensate for PORV unavailability.

I.1 and I.2

With the pressurizer level above [220] inches and the PORV inoperable or the LTOP System inoperable for any reason other than cited in Condition A through H, Required Action I.2 requires the RCS depressurized and vented within 12 hours from the time either Condition started.

One or more vents may be used. A vent size of ≥ [0.75] square inches is specified. This vent size assumes 100 psig backpressure. Because makeup may be required, the vent size accommodates inadvertent full makeup system operation. Such a vent keeps the pressure from full flow of [one] makeup pump with a wide open makeup control valve within the LCO limit.

The PORV has a larger area and may be used for venting by opening and locking it open.

This size RCS vent or the PORVs a vent cannot maintain RCS pressure below LTOP limits if the HPI and CFT systems are inadvertently actuated. Therefore, verification of the deactivation of two HPI pumps, HPI injection, and the CFTs must accompany the depressurizing and venting. Since these systems are required deactivated by the LCO, SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 require verification of their deactivated status every 12 hours.

The Completion Time is based on operating experience that this activity can be accomplished in this time period and on engineering evaluations indicating that a limiting LTOP transient is not likely in this time.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

Verifications must be performed that only [one] makeup pump is capable of injecting into the RCS, the HPI is deactivated, and the CFT discharge isolation valves are closed and immobilized. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP System. The Surveillances are required at 12 hour intervals.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The 12 hour intervals are shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation within the safety analysis.

SR 3.4.12.4

Verification of the pressurizer level at ≤ [220] inches by observing control room or other indications ensures a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients.

The 30 minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level variations. This Frequency may be discontinued when the ends of these conditions are satisfied, as defined in plant procedures. Thereafter, the Surveillance is required at 12 hour intervals.

These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

SR 3.4.12.5

Verification that the PORV block valve is open ensures a flow path to the PORV. This is required at 12 hour intervals.

The interval has been shown by operating practice sufficient to regularly assess conditions for potential degradation and verify operation is within the safety analysis.

SR 3.4.12.6

The RCS vent of at least [0.75] square inches must be verified open for relief protection only if the vent is being used to satisfy the requirements of this LCO. For a vent valve not locked open, the Frequency is every 12 hours. Valves that are sealed or secured in the open position are considered "locked" in this context. For other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position, a removed pressurizer safety valve, or open manway), the required Frequency is every 31 days.

Again, the Frequency intervals consider operating practice to determine adequacy to regularly assess conditions for potential degradation and verify operation within the safety analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The passive vent path arrangement must only be open to be OPERABLE.

A Note modifies the SR by requiring the Surveillance to only be met when complying with LCO 3.4.12.b.

SR 3.4.12.7

A CHANNEL FUNCTIONAL TEST is required within [12] hours after decreasing RCS temperature to ≤ [283]°F and every 31 days thereafter to ensure the setpoint is proper for using the PORV for LTOP. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. PORV actuation is not needed, as it could depressurize the RCS.

The [12] hour Frequency considers the unlikelihood of a low temperature overpressure event during the time. The 31 day Frequency is based on industry accepted practice and is acceptable by experience with equipment reliability.

SR 3.4.12.8

The performance of a CHANNEL CALIBRATION is required every [18] months. The CHANNEL CALIBRATION for the LTOP setpoint ensures that the PORV will be actuated at the appropriate RCS pressure by verifying the accuracy of the instrument string. The calibration can only be performed in shutdown.

The Frequency considers a typical refueling cycle and industry accepted practice.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. FSAR, Section 15.
4. 10 CFR 50.46.

BASES

REFERENCES (continued)

5. 10 CFR 50, Appendix K.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting Leakage Detection Systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area are 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 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 analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA). However, the ability to monitor leakage provides advance warning to permit plant shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

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 1 gpm 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 leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100.

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

   No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

   One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

BASES

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 or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

[e. Primary to Secondary LEAKAGE through Any One SG

The [720] gallon per day limit on one SG allocates the total 1 gpm allowed primary to secondary LEAKAGE equally between the two generators.]

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.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

BASES

ACTIONS

A.1

If unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE are in excess of the LCO limits, the LEAKAGE 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 if unidentified, identified, or primary to secondary 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. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The Completion Times allowed are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE within the LCO limits ensures that 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 an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, [and RCP seal injection and return flows]). Therefore, a Note is added allowing 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP pump seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Chapter [15].

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the RCS pressure boundary that separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt.

A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Decay Heat Removal (DHR) System,

BASES

BACKGROUND (continued)

b. Emergency Core Cooling System (ECCS), and

c. Makeup and Purification System.

The PIVs are listed in [FSAR section] Reference 6.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the DHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the reactor coolant pressure boundary (RCPB), and the subsequent pressurization of the DHR System downstream of the PIVs from the RCS. Because the low pressure portion of the DHR System is typically designed for 600 psig, overpressurization failure of the DHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage 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 on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

BASES

LCO (continued)

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the DHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the DHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

<u>A.1 and A.2</u>

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be on the RCS pressure boundary [or the high pressure portion of the system].

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hours allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or

BASES

ACTIONS (continued)

restoring one leaking PIV. [The 72 hour time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

or

The 72 hour time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period.]

- REVIEWER'S NOTE -

Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.

B.1 and B.2

If leakage cannot be reduced, [the system isolated,] or 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 to MODE 5 within 36 hours. This Required Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

The inoperability of the DHR autoclosure interlock renders the DHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the DHR systems design pressure. If the DHR autoclosure interlock is inoperable, operation may continue as long as the DHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This action accomplishes the purpose of the autoclosure function.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 or A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every [18] months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The [18 month] Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 7), and is based on the need to perform such surveillances under conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the plant at power.

[In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.]

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the DHR System when the DHR System is aligned to

BASES

SURVEILLANCE REQUIREMENTS (continued)

the RCS in the decay heat removal mode of operation. PIVs contained in the DHR flow path must be leakage rate tested after DHR is secured and stable unit conditions and the necessary differential pressures are established.

- REVIEWER'S NOTE -

The "24 hour..." Frequency of performance for Surveillance Requirement 3.4.14.1 is not required for B&W Owner's Group plants licensed prior to 1980. These plants were licensed prior to the NRC establishing formal Technical Specification controls for pressure isolation valves. Subsequently, these earlier plants had their [licenses modified by NRC Order to require certain PIV testing Frequencies (excluding the "24 hour..." Frequency) be included in that plant's Technical Specifications. Based upon the information available to the Staff at the time, the content of those Orders was considered acceptable. Since 1980, the NRC Staff has determined an additional PIV leakage rate determination is required within 24 hours following actuation of the valve and flow through the valve. This is necessary in order to ensure the PIV's ability to support the integrity of the reactor coolant pressure boundary. The Revised Standard Technical Specifications include the "24 hours..." Frequency to reflect current NRC Staff position on the need to include this test requirement within Technical Specifications.

[SR 3.4.14.2 and SR 3.4.14.3

Verifying that the DHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not pressurize the DHR system beyond 125% of its design pressure of [600] psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < [425] psig to open the valves. This setpoint ensures the DHR design pressure will not be exceeded and the DHR relief valves will not lift. The 18 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 was performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

These SRs are modified by Notes allowing the DHR autoclosure function to be disabled when using the DHR System suction relief valve for cold overpressure protection in accordance with LCO 3.4.12.]

BASES

REFERENCES

1. 10 CFR 50.2.

2. 10 CFR 55a(c).

3. 10 CFR 50, Appendix A, Section V, GDC 55.

4. NUREG-75/014, Appendix V, October 1975.

5. NUREG-0677, NRC, May 1980.

6. [Document containing list of PIVs.]

7. ASME, Boiler and Pressure Vessel Code, Section XI.

8. 10 CFR 50.55a(g).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can readily be 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 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} μCi/cc radioactivity for particulate monitoring and of 10^{-6} μCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew point is well within the sensitivity range of available instruments.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid

BASES

BACKGROUND (continued)

flow into or from the containment sump [and condensate flow from air coolers]. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required for this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS LEAKAGE into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 3). Multiple instrument locations are utilized, if needed, to ensure the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area are 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 detrimental to the safety of the unit and the public.

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 extremely small leaks are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

BASES

LCO (continued)

The LCO requirements are satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor, in combination with a particulate or gaseous radioactivity monitor, provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

The Actions are modified by a Note indicating that the provisions of LCO 3.0.4 do not apply. As a result, a MODE change is allowed when the sump and required radiation monitors are inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.

<u>A.1 and A.2</u>

With the required containment sump monitor inoperable, no other form of sampling can provide the equivalent information.

However, the containment atmosphere activity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS inventory balance, SR 3.4.13.1, water inventory balance, 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 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and [RCP seal injection and return flows]). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required sump monitor to OPERABLE status is required to regain the function in a Completion Time of 30 days after the monitor's failure. This time is acceptable considering the frequency and

BASES

ACTIONS (continued)

adequacy of the RCS water inventory balance required by Required Action A.1.

B.1.1, B.1.2, and B.2

With required gaseous or particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. With a sample obtained and analyzed or a water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors.

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and [RCP seal injection and return flows]). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leak detection is available.

C.1 and C.2

If a Required Action of Condition A or B cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

With both required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that each channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitor. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. 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 instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the required RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45.
3. FSAR, Section [ ].

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The Code of Federal Regulations, 10 CFR 100 (Ref. 1), specifies the maximum dose to the whole body and the thyroid an individual at the site boundary can receive for 2 hours during an accident. The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for an SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits (Ref. 1). Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following an SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limits and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The analysis also assumes a reactor trip and a turbine trip at the same time as the SGTR event.

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the facility that could affect RCS specific activity as they relate to the acceptance limits.

The rise in pressure in the ruptured SG causes radioactively contaminated steam to discharge to the atmosphere through the

BASES

APPLICABLE SAFETY ANALYSES (continued)

atmospheric dump valves or the main steam safety valves. The atmospheric discharge stops when the turbine bypass to the condenser removes the excess energy to rapidly reduce the RCS pressure and close the valves. The unaffected SG removes core decay heat by venting steam until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable Specification, for more than 48 hours.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of an SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

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 gross specific activity in the primary coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

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 gross specific activity are necessary to contain the potential

BASES

APPLICABILITY (continued)

consequences of an SGTR to within the acceptable 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 an SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the atmospheric dump valves and 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 demonstrate the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling must continue for trending.

The DOSE EQUIVALENT I-131 must be restored to limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note to the Required Actions 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

If a Required Action and associated Completion Time of Condition A are not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is required to get to MODE 3 below 500°F without challenging reactor emergency systems.

C.1

With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.

BASES

ACTIONS (continued)

The allowed Completion Time of 6 hours to reach MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves, and prevents venting the SG to the environment in an SGTR event. The Completion Time of 6 hours is required to reach MODE 3 from full power conditions in an orderly manner and without challenging reactor emergency systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once per 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with RCS average temperature at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during that time period.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure the iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross specific activity is monitored every 7 days. 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 failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

SR 3.4.16.3 requires radiochemical analysis for $\bar{E}$ determination every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specific gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies

BASES

SURVEILLANCE REQUIREMENTS (continued)

per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that requires sampling to be performed 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 100.11.

2. FSAR, Section [15.6.3].

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Core Flood Tanks (CFTs)

BASES

BACKGROUND

The function of the ECCS CFTs is to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA. Two CFTs are provided for these functions.

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 follows immediately, reactor coolant inventory has vacated the core through steam flashing and ejection through the break. The core is essentially in adiabatic heatup. The balance of inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection water.

The CFTs are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The CFTs are passive components, since no operator or control actions are required for them to perform their function. Internal tank pressure is sufficient to discharge the contents of the CFTs to the RCS if RCS pressure decreases below the CFT pressure. Each CFT is piped separately into the reactor vessel downcomer. The CFT injection lines are also utilized by the Low Pressure Injection (LPI) System. Each CFT is isolated from the RCS by a motor operated isolation valve and two check valves in series.

The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident.

The CFTs thus form a passive system for injection directly into the reactor vessel. Except for the core flood line break LOCA, a unique accident that also disables a portion of the injection system, both tanks are assumed to operate in the safety analyses for Design Basis Events. Because injection is directly into the reactor vessel downcomer, and

BASES

BACKGROUND (continued)

because it is a passive system not subject to the single active failure criterion, all fluid injection is credited for core cooling.

The CFT gas/water volumes, gas pressure, and outlet pipe size are selected to provide core cooling for a large break LOCA prior to the injection of coolant by the LPI System.

APPLICABLE SAFETY ANALYSES

The CFTs are taken credit for in both the large and small break LOCA analyses at full power (Ref. 1). These Design Basis Accident (DBA) analyses establish the acceptance limits for the CFTs. Reference to the analyses for these DBAs is used to assess changes in the CFTs as they relate to the acceptance limits. In performing the LOCA calculations, conservative assumptions are made concerning the availability of emergency injection flow. The assumption of the loss of offsite power is required by regulations. In the early stages of a LOCA with the loss of offsite power, the CFTs provide the sole source of makeup water to the RCS.

This is because the LPI pumps and high pressure injection (HPI) pumps cannot deliver flow until the emergency diesel generators (EDGs) start, come to rated speed, and go through their timed loading sequence.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

During this event, the CFTs discharge to the RCS as soon as RCS pressure decreases below CFT pressure. As a conservative estimate, no credit is taken for HPI for large break LOCAs. LPI is not assumed to occur until 35 seconds after the RCS pressure decreases to the ESFAS actuation pressure. No operator action is assumed during the blowdown stage of a large break LOCA.

The small break LOCA analysis also assumes a time delay after ESFAS actuation before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated by the CFTs, with pumped flow then providing continued cooling. As break size decreases, the CFTs and HPI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the CFTs continues to decrease until the tanks are not required and the HPI pumps become responsible for terminating the temperature increase.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This LCO helps to ensure that the following acceptance criteria for the ECCS established by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature of 2200°F,

b. Maximum cladding oxidation of ≤ 0.17 times the total cladding thickness before oxidation,

c. Maximum hydrogen generation from a zirconium water reaction of ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react, and

d. Core maintained in a coolable geometry.

Since the CFTs discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

The limits for operation with a CFT that is inoperable for any reason other than the boron concentration not being within limits minimize the time that the plant is exposed to a LOCA event occurring along with failure of a CFT, which might result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened, or the proper water volume or nitrogen cover pressure cannot be restored, the full capability of one CFT is not available and prompt action is required to place the reactor in a MODE in which this capability is not required.

In addition to LOCA analyses, the CFTs have been assumed to operate to provide borated water for reactivity control for severe overcooling events such as a large steam line break (SLB).

The CFTs are part of the primary success path that functions or actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The minimum volume requirement for the CFTs ensures that both CFTs can provide adequate inventory to reflood the core and downcomer following a LOCA. The downcomer then remains flooded until the HPI and LPI systems start to deliver flow.

The maximum volume limit is based upon the need to maintain adequate gas volume to ensure proper injection, ensure the ability of the CFTs to

BASES

APPLICABLE SAFETY ANALYSES (continued)

fully discharge, and limit the maximum amount of boron inventory in the CFTs. Values of [7555] gallons and [8005] gallons are specified. These values allow for instrument inaccuracies. Values of other parameters are treated similarly.

The minimum nitrogen cover pressure requirement of [525] psig ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analysis.

The maximum nitrogen cover pressure limit of [625] psig ensures that the amount of CFT inventory that is discharged while the RCS depressurizes, and is therefore lost through the break, will not be larger than that predicted by the safety analysis. The maximum allowable boron concentration of [3500] ppm in the CFTs ensures that the sump pH will be maintained between 7.0 and 11.0 following a LOCA.

The minimum boron requirement of [2270] ppm is selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. During a large break LOCA, all control rod assemblies are assumed not to insert into the core, and the initial reactor shutdown is accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the CFTs to prevent a return to criticality during reflood.

The CFT isolation valves are not single failure proof; therefore, whenever these valves are open, power shall be removed from them. This precaution ensures that both CFTs are available during an accident. With power supplied to the valves, a single active failure could result in a valve closure, which would render one CFT unavailable for injection. Both CFTs are required to function in the event of a large break LOCA.

The CFTs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the CFTs are available to accomplish their core cooling safety function following a LOCA. Both CFTs are required to function in the event of a large break LOCA. If the entire contents of both tanks are not injected during the blowdown phase of a large break LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated. For a CFT to be considered OPERABLE, the isolation valve must be fully open, power removed above [2000] psig, and the limits established in the SR for contained volume, boron concentration, and nitrogen cover pressure must be met.

BASES

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 750 psig, the CFT OPERABILITY requirements are based on full power operation. Although cooling requirements may decrease as power decreases, the CFTs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures ≥ 750 psig. Below 750 psig, the rate of RCS blowdown is such that the safety injection pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3 with RCS pressure ≤ 750 psig, and in MODES 4, 5, and 6, the CFT motor operated isolation valves are closed to isolate the CFTs from the RCS. This allows RCS cooldown and depressurization without discharging the CFTs into the RCS or requiring depressurization of the CFTs.

ACTIONS

A.1

If the boron concentration of one CFT is not within limits, it must be returned to within the limits within 72 hours. In this condition, ability to maintain subcriticality may be reduced, but the effects of reduced boron concentration on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the CFT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of two CFTs, the consequences are less severe than they would be if the contents of a CFT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one CFT is inoperable for a reason other than boron concentration, the CFT must be returned to OPERABLE status within 1 hour. In this condition it cannot be assumed that the CFT will perform its required function during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable CFT to OPERABLE status. The Completion Time minimizes the time the plant is potentially exposed to a LOCA in these conditions.

BASES

ACTIONS (continued)

C.1 and C.2

If the CFT cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and RCS pressure reduced to ≤ 750 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 CFT is inoperable, the unit is in a condition outside the accident analysis; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification every 12 hours that each CFT isolation valve is fully open, as indicated in the control room, ensures that the CFTs are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in accident analysis assumptions not being met. A 12 hour Frequency is considered reasonable in view of administrative controls that ensure that a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Verification every 12 hours of each CFT's nitrogen cover pressure and the borated water volume is sufficient to ensure adequate injection during a LOCA. Due to the static design of the CFTs, a 12 hour Frequency usually allows the operator to identify changes before the limits are reached. Operating experience has shown that this Frequency is appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

Surveillance once every 31 days is reasonable to verify that the CFT boron concentration is within the required limits, because the static design of the CFT limits the ways in which the concentration can be changed. The Frequency is adequate to identify changes that could

BASES

SURVEILLANCE REQUIREMENTS (continued)

occur from mechanisms such as stratification or inleakage. Sampling within 6 hours after an 80 gallon volume increase will identify whether inleakage from the RCS has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the borated water storage tank (BWST), because the water contained in the BWST is within CFT boron concentration requirements. This is consistent with the recommendations of NUREG-1366 (Ref. 3).

SR 3.5.1.5

Verification every 31 days that power is removed from each CFT isolation valve operator [when the RCS pressure is ≥ [2000] psig ensures that an active failure could not result in the undetected closure of a CFT motor operated isolation valve coincident with a LOCA. If this closure were to occur and the postulated LOCA is a rupture of the redundant CFT inlet piping, CFT capability would be rendered inoperable. The rupture would render the tank with the open valve inoperable, and a closed valve on the other CFT would likewise render it inoperable. This would cause a loss of function for the CFTs. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that the power is removed.

This SR is modified by a Note that allows power to be supplied to the motor operated isolation valves when RCS pressure is < [2000] psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

REFERENCES

1. FSAR, Section [6.3].
2. 10 CFR 50.46.
3. Draft NUREG-1366, February 1990.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA),

b. Rod ejection accident (REA),

c. Steam generator tube rupture (SGTR), and

d. Steam line break (SLB).

There are two phases of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the Reactor Coolant System (RCS) via the cold legs and to the reactor vessel. After the borated water storage tank (BWST) has been depleted, the ECCS recirculation phase is entered as the ECCS suction is transferred to the containment sump.

Two redundant, 100% capacity trains are provided. In MODES 1, 2, and 3, each train consists of high pressure injection (HPI) and low pressure injection (LPI) subsystems. In MODES 1, 2, and 3, both trains must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided even in the event of a single active failure.

A suction header supplies water from the BWST or the containment sump to the ECCS pumps. Separate piping supplies each train. HPI discharges into each of the four RCS cold legs between the reactor coolant pump and the reactor vessel. LPI discharges into each of the two core flood nozzles on the reactor vessel that discharge into the vessel downcomer area. Control valves are set to balance the HPI flow to the RCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a small break LOCA in one of the RCS cold legs near an HPI nozzle.

The HPI pumps are capable of discharging to the RCS at an RCS pressure above the opening setpoint of the pressurizer safety valves. The LPI pumps are capable of discharging to the RCS at an RCS pressure of approximately 200 psia. When the BWST has been nearly emptied, the suction for the LPI pumps is manually transferred to the containment sump. The HPI pumps cannot take suction directly from the

BASES

BACKGROUND (continued)

sump. If HPI is still needed, a cross connect from the discharge side of the LPI pump to the suction of the HPI pumps would be opened. This is known as "piggy backing" HPI to LPI and enables continued HPI to the RCS, if needed, after the BWST is emptied.

In the long term cooling period, flow paths in the LPI System are established to preclude the possibility of boric acid in the core region reaching an unacceptably high concentration. One flow path is from the hot leg through the decay heat suction line from the hot leg and then in a reverse direction through the containment sump outlet line into the sump. The other flow path is through the pressurizer auxiliary spray line from one LPI train into the pressurizer and through the hot leg into the top region of the core.

The HPI subsystem also functions to supply borated water to the reactor core following increased heat removal events, such as large SLBs.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

During a large break LOCA, RCS pressure will decrease to < 200 psia in < 20 seconds. The ECCS is actuated upon receipt of an Engineered Safety Feature Actuation System (ESFAS) signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately (in the programmed sequence). If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the diesel generators. Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive core flood tanks (CFTs) and the BWST covered in LCO 3.5.1, "Core Flood Tanks (CFTs)," and LCO 3.5.4, "Borated Water Storage Tank (BWST)," provide the cooling water necessary to meet 10 CFR 50.46 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 1), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F,

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

d. Core is maintained in a coolable geometry, and

e. Adequate long term core cooling capability is maintained.

The LCO also helps ensure that containment temperature limits are met.

Both HPI and LPI subsystems are assumed to be OPERABLE in the large break LOCA analysis at full power (Ref. 2). This analysis establishes a minimum required flow for the HPI and LPI pumps, as well as the minimum required response time for their actuation. The HPI pump is credited in the small break LOCA analysis. This analysis establishes the flow and discharge head requirements at the design point for the HPI pump. The SGTR and SLB analyses also credit the HPI pump but are not limiting in their design.

The large break LOCA event with a loss of offsite power and a single failure (disabling one ECCS train) establishes the OPERABILITY requirements for the ECCS. During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or CONTROL ROD assembly insertion for small breaks. Following depressurization, emergency cooling water is injected into the reactor vessel core flood nozzles, then flows into the downcomer, fills the lower plenum, and refloods the core.

The LCO ensures that an ECCS train will deliver sufficient water to match decay heat boiloff rates soon enough to minimize core uncovery for a large break LOCA. It also ensures that the HPI pump will deliver sufficient water for a small break LOCA and provide sufficient boron to maintain the core subcritical.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In the LOCA analyses, HPI and LPI are not credited until 35 seconds after actuation of the ESFAS signal. This is based on a loss of offsite power and the associated time delays in startup and loading of the emergency diesel generator (EDG). Further, LPI flow is not credited until RCS pressure drops below the pump's shutoff head. For a large break LOCA, HPI is not credited at all.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that at least one is available, assuming a single failure in the other train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of an HPI subsystem and an LPI subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ESFAS signal and manually transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the BWST to the RCS via the HPI and LPI pumps and their respective discharge flow paths to each of the four cold leg injection nozzles and the reactor vessel. In the long term, this flow path may be manually transferred to take its supply from the containment sump and to supply its flow to the RCS via two paths, as described in the Background section.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

As indicated in the Note, operation in MODE 3 with ECCS trains de-activated pursuant to LCO 3.4.12 is necessary for plants with an LTOP System arming temperature at or near the MODE 3 boundary temperature of [350]°F. LCO 3.4.12 requires that certain components be de-activated at and below the LTOP System arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the systems to OPERABLE status.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the ECCS train OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPI pump performance is based on the small break LOCA, which establishes the pump performance curve and is less dependent on power. The HPI pump performance requirements are based on a small break LOCA. MODES 2 and 3 requirements are bounded by the MODE 1 analysis.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one or more trains operable and at least 100% of the injection flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 3) that are based on a risk evaluation and is a reasonable time for many repairs.

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 3) has shown the risk of having one full ECCS train

BASES

ACTIONS (continued)

inoperable to be sufficiently low to justify continued operation for 72 hours.

With one or more components inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable components cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and at least 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.

C.1

Condition A is applicable with one or more trains inoperable. The allowed Completion Time is based on the assumption that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the facility is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power or by key locking the control in the correct position ensures that the valves cannot change position as the result of an active failure. These valves are of the type described in Reference 4, which can disable the function of both ECCS trains and invalidate the accident analyses. The 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure the unlikelihood of a mispositioned valve.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an inoperable valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

With the exception of systems in operation, the ECCS pumps are normally in a standby, nonoperating mode. As such, the flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an ESFAS signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the existence of procedural controls governing system operation.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code (Ref. 5). This type of testing may be accomplished by measuring the pump's developed head at only one point of the pump's characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant accident analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the

BASES

SURVEILLANCE REQUIREMENTS (continued)

ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These SRs demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated ESFAS signal and that each ECCS pump starts on receipt of an actual or simulated ESFAS signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 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. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ESFAS testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

This Surveillance ensures that these valves are in the proper position to prevent the HPI pump from exceeding its runout limit. This 18 month Frequency is based on the same reasons as those stated for SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

This Surveillance ensures that the flow controllers for the LPI throttle valves will automatically control the LPI train flow rate in the desired range and prevent LPI pump runout as RCS pressure decreases after a LOCA. The 18 month Frequency is justified by the same reasons as those stated for SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.9

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to preserve access to the location, and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This

BASES

SURVEILLANCE REQUIREMENTS (continued)

Frequency has been found to be sufficient to detect abnormal degradation and has been confirmed by operating experience.

REFERENCES

1. 10 CFR 50.46.
2. FSAR, Section [6.3].
3. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
4. IE Information Notice 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," January 6, 1987.
5. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

The Background section for Bases B 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: high pressure injection (HPI) and low pressure injection (LPI), each consisting of two redundant, 100% capacity trains.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps, such that water from the borated water storage tank (BWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 is applicable to these Bases.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. Included in these reductions is that certain automatic Engineered Safety Feature Actuation System (ESFAS) actuation is not available. In this MODE sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one ECCS train is required for MODE 4. This requirement dictates that single failures are not considered during this MODE. The ECCS train - shutdown satisfies Criterion 3 of the 10 CFR 50.36(c)(2(ii).

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to ensure sufficient ECCS flow is available to the core following a DBA.

In MODE 4, an ECCS train consists of an HPI subsystem and an LPI subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the BWST to the RCS, via the ECCS pumps and their respective supply headers, to each of the four

BASES

LCO (continued)

cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS hot and cold legs.

This LCO is modified by two Notes. The first allows a DHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the DHR mode during MODE 4. The second Note states that HPI actuation may be deactivated in accordance with LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." Operator action is then required to initiate HPI. In the event of a loss of coolant accident (LOCA) requiring HPI actuation, the time required for operator action has been shown by analysis to be acceptable.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for the ECCS are covered by LCO 3.5.2.

In MODE 4 with the RCS temperature below 280°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "DHR and Coolant Circulation - High Water Level," and LCO 3.9.5, "DHR and Coolant Circulation - Low Water Level."

ACTIONS

A.1

If no LPI subsystem train is OPERABLE, the unit is not prepared to respond to a LOCA or to continue cooldown using the LPI pumps and decay heat exchangers. The Completion Time of immediately, which would initiate action to restore at least one ECCS LPI subsystem to OPERABLE status, ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat must be removed by an LPI train operating with suction from the RCS. If no LPI train is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam

BASES

ACTIONS (continued)

generator(s). The alternate means of heat removal must continue until the inoperable ECCS LPI subsystem can be restored to operation so that continuation of decay heat removal (DHR) is provided.

With both DHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the LPI trains operating in the DHR mode. Therefore, the appropriate action is to initiate measures to restore one ECCS LPI subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

B.1

If no ECCS HPI subsystem is OPERABLE, due to the inoperability of the HPI pump or flow path from the BWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring ESFAS. The 1 hour Completion Time to restore at least one ECCS HPI subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

C.1

When the Required Action of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. The allowed Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

None.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Borated Water Storage Tank (BWST)

BASES

BACKGROUND

The BWST supports the ECCS and the Containment Spray System by providing a source of borated water for ECCS and containment spray pump operation. In addition, the BWST supplies borated water to the refueling pool for refueling operations.

The BWST supplies two ECCS trains, each by a separate, redundant supply header. Each header also supplies one train of the Containment Spray System. A normally open, motor operated isolation valve is provided in each header to allow the operator to isolate the BWST from the ECCS after the ECCS pump suction has been transferred to the containment sump following depletion of the BWST during a loss of coolant accident (LOCA). Use of a single BWST to supply both ECCS trains is acceptable because the BWST is a passive component, and passive failures are not assumed in the analysis of Design Basis Events (DBEs) to occur coincidentally with the Design Basis Accident (DBA).

The ECCS and containment spray pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at shutoff head conditions.

This LCO ensures that:

a. The BWST contains sufficient borated water to support the ECCS during the injection phase,

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and containment spray pumps at the time of transfer to the recirculation mode of cooling, and

c. The reactor remains subcritical following a LOCA.

Insufficient water inventory in the BWST could result in insufficient cooling capacity of the ECCS when the transfer to the recirculation mode occurs.

Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside containment.

BASES

APPLICABLE SAFETY ANALYSES

During accident conditions, the BWST provides a source of borated water to the high pressure injection (HPI), low pressure injection (LPI), and containment spray pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown. The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of Specifications B 3.5.2, "ECCS - Operating," and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the BWST in order to evaluate their effects in relation to the acceptance limits.

The limits on volume of [≥ 415,200 gallons and ≤ 449,000 gallons] are based on several factors. Sufficient deliverable volume must be available to provide at least 20 minutes of full flow of all ECCS pumps prior to the transfer to the containment sump for recirculation. Twenty minutes gives the operator adequate time to prepare for switchover to containment sump recirculation.

A second factor that affects the minimum required BWST volume is the ability to support continued ECCS pump operation after the manual transfer to recirculation occurs. When ECCS pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head (NPSH) for the LPI and containment spray pumps. This NPSH calculation is described in the FSAR (Ref. 1), and the amount of water that enters the sump from the BWST and other sources is one of the input assumptions. Since the BWST is the main source that contributes to the amount of water in the sump following a LOCA, the calculation does not take credit for more than the minimum volume of usable water from the BWST.

The third factor is that the volume of water in the BWST must be within a range that will ensure the solution in the sump following a LOCA is within a specified pH range that will minimize the evolution of iodine and the effect of chloride and caustic stress corrosion cracking on the mechanical systems and components.

The volume range ensures that refueling requirements are met and that the capacity of the BWST is not exceeded. Note that the volume limits refer to total, rather than usable, volume required to be in the BWST; a certain amount of water is unusable because of tank discharge line location or other physical characteristics.

The [2270] ppm limit for minimum boron concentration was established to ensure that, following a LOCA, with a minimum BWST level, the reactor

BASES

APPLICABLE SAFETY ANALYSES (continued)

will remain subcritical in the cold condition following mixing of the BWST and Reactor Coolant System (RCS) water volumes. Large break LOCAs assume that all control rods remain withdrawn from the core.

The minimum and maximum concentration limits both ensure that the solution in the sump following a LOCA is within a specified pH range that will minimize the evolution of iodine and the effect of chloride and caustic stress corrosion cracking on the mechanical systems and components.

The [2450] ppm maximum limit for boron concentration in the BWST is also based on the potential for boron precipitation in the core during the long term cooling period following a LOCA. For a cold leg break, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point may be reached where boron precipitation will occur in the core. Post LOCA emergency procedures direct the operator to establish dilution flow paths in the LPI System to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA.

Boron concentrations in the BWST in excess of the limit could result in precipitation earlier than assumed in the analysis.

The 40°F lower limit on the temperature of the solution in the BWST was established to ensure that the solution will not freeze. This temperature also helps prevent boron precipitation and ensures that water injection in the reactor vessel will not be colder than the lowest temperature assumed in reactor vessel stress analysis. The [100]°F upper limit on the temperature of the BWST contents is consistent with the maximum injection water temperature assumed in the LOCA analysis.

The numerical values of the parameters stated in the SR are actual values and do not include allowance for instrument errors.

The BWST satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The BWST exists to ensure that an adequate supply of borated water is available to cool and depressurize the containment in the event of a DBA; to cool and cover the core in the event of a LOCA, thereby ensuring the reactor remains subcritical following a DBA; and to ensure an adequate

BASES

LCO (continued)

level exists in the containment sump to support ECCS and containment spray pump operation in the recirculation MODE. To be considered OPERABLE, the BWST must meet the limits for water volume, boron concentration, and temperature established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, the BWST OPERABILITY requirements are dictated by the ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the BWST must be OPERABLE to support their operation.

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled," respectively. MODE 6 core cooling requirements are addressed by LCO 3.9.4, "DHR and Coolant Circulation - High Water Level," and LCO 3.9.5, "DHR and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With either the BWST boron concentration or borated water temperature not within limits, the condition must be corrected within 8 hours. In this condition, neither the ECCS nor the Reactor Building Spray System can perform its design functions. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which these systems are not required. The 8 hour limit to restore the temperature or boron concentration to within limits was developed considering the time required to change boron concentration or temperature and assuming that the contents of the tank are still available for injection.

B.1

With the BWST inoperable for reasons other than Condition A (e.g., water volume), levels must be restored to within required limits within 1 hour. In this condition, neither the ECCS nor the Containment Spray System can perform its design functions. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the BWST is not required. The allowed Completion Time of 1 hour to restore the BWST to OPERABLE status is based on this condition simultaneously affecting multiple redundant trains.

BASES

ACTIONS (continued)

C.1 and C.2

If the BWST cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

Verification every 24 hours that the BWST water temperature is within the specified temperature band ensures that the boron will not precipitate; the fluid will not freeze; the fluid temperature entering the reactor vessel will not be colder than assumed in the reactor vessel stress analysis; and the fluid temperature entering the reactor vessel will not be hotter than assumed in the LOCA analysis. The 24 hour Frequency is sufficient to identify a temperature change that would approach either temperature limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that requires the Surveillance to be performed only when ambient air temperatures are outside the operating temperature limits of the BWST. With ambient temperatures within this band, the BWST temperature should not exceed the limits.

SR 3.5.4.2

Verification every 7 days that the BWST contained volume is within the required range ensures that a sufficient initial supply is available for injection and to support continued ECCS pump operation on recirculation. Since the BWST volume is normally stable and provided with a low level alarm, a 7 day Frequency has been shown to be appropriate through operating experience.

SR 3.5.4.3

Verification every 7 days that the boron concentration of the BWST fluid is within the required band ensures that the reactor will remain subcritical following a LOCA. Since the BWST volume is normally stable, a 7 day sampling Frequency is appropriate and has been shown to be acceptable through operating experience.

BASES

REFERENCES	1.	FSAR, Section [6.1].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building (RB), its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. For containments with ungrouted tendons, the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed using a three way post tensioning system. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The concrete RB is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option [A][B] (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

    1. Capable of being closed by an OPERABLE automatic containment isolation system or

    2. Closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves,"

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks,"

BASES

BACKGROUND (continued)

c. All equipment hatches are closed, and

[d. The pressurized sealing mechanism associated with each penetration, except as provided in LCO 3.6.[ ], is OPERABLE.]

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 rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a 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. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.25]% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option [A][B] (Ref. 1), as L_a: the maximum allowable leakage rate at the calculated maximum peak containment pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be [0.25]% per day in the safety analysis at P_a = [53.9] psig (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(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. 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) [and purge valves with resilient seals (LCO 3.6.3)] are not specifically part

BASES

LCO (continued)

of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 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. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures 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.

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 and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B,

BASES

SURVEILLANCE REQUIREMENTS (continued)

and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage, and [$< 0.75\ L_a$ for Option A] [$\leq 0.75\ L_a$ for Option B] 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.

- REVIEWER'S NOTE -

Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

[SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 4).]

REFERENCES

1. 10 CFR 50, Appendix J, Option [A][B].
2. FSAR, Sections [14.1 and 14.2].
3. FSAR, Section [5.6].
4. Regulatory Guide 1.35, Revision [1].

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 ft 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 is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock door is provided with limit switches that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.25]% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option A (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated

BASES

APPLICABLE SAFETY ANALYSES (continued)

maximum peak containment pressure (P_a) following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock. L_a is [0.25]% per day and P_a is [53.9] psig, resulting from the limiting design basis LOCA.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As a part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this 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

BASES

ACTIONS (continued)

containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock.

This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the remaining OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered 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 based on

BASES

ACTIONS (continued)

engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

<u>B.1, B.2, and B.3</u>

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from the

BASES

ACTIONS (continued)

containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated 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. This action must be completed 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

BASES

ACTIONS (continued)

plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable, since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 the 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.

REFERENCES

1. 10 CFR 50, Appendix J, Option [A][B].
2. FSAR, Sections [14.1 and 14.2].
3. FSAR, Section [5.6].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on an automatic isolation signal. These isolation devices consist of either passive devices or active (automatic) devices. 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 following an accident without operator action, 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.

Containment isolation occurs upon receipt of a high containment pressure or diverse containment isolation signal. The containment isolation signal closes automatic containment isolation valves in fluid penetrations not required for operation of engineered safeguard systems to prevent leakage of radioactive material. Upon actuation of high pressure injection, automatic containment valves also isolate systems not required for containment or Reactor Coolant System (RCS) heat removal. Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated in the event of a release of radioactive material to containment atmosphere from the RCS following a Design Basis Accident (DBA).

OPERABILITY of the containment isolation valves (and blind flanges) supports containment OPERABILITY during accident conditions.

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 the containment function assumed in the safety analysis will be maintained.

The Reactor Building Purge System is part of the Reactor Building Ventilation System. The Purge System was designed for intermittent

BASES

BACKGROUND (continued)

operation, providing a means of removing airborne radioactivity caused by minor leakage from the RCS prior to personnel entry into containment. The Containment Purge System consists of one [48] inch line for exhaust and one [48] inch line for supply, with supply and exhaust fans capable of purging the containment atmosphere at a rate of approximately [50,000] ft^3/min. This flow rate is sufficient to reduce the airborne radioactivity level within containment to levels defined in 10 CFR 20 (Ref. 1) for a 40 hour workweek within 2 hours of purge initiation during reactor operation. The containment purge supply and exhaust lines each contain two isolation valves that receive an isolation signal on a unit vent high radiation condition.

Failure of the purge valves to close following a design basis event would cause a significant increase in the radioactive release because of the large containment leakage path introduced by these [48] inch purge lines. Failure of the purge valves to close would result in leakage considerably in excess of the containment design leakage rate of [0.25]% of containment air weight per day (L_a) (Ref. 2). Because of their large size, the [48] inch purge valves in some units are not qualified for automatic closure from their open position under DBA conditions. Therefore, the [48] inch purge valves are maintained sealed closed (SR 3.6.3.1) in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

The [8 inch] containment minipurge valves operate to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access and

b. Equalize internal and external pressures.

Since the minipurge valves 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 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), a main steam line break, and a rod ejection accident (Ref. 3). In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analysis assumes that the [48] inch purge valves are closed at event initiation.

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

The single-failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the containment purge valves. Two valves in a 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 provided with diverse power sources, motor operated and pneumatically operated spring closed, respectively. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.

The purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single-failure criterion remains applicable to the containment purge valves because of failure in the control circuit associated with each valve. Again, the purge system valve design prevents a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve 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 [48] inch purge valves must be maintained sealed closed [or have blocks installed to prevent full opening]. [Blocked purge valves also actuate on an automatic signal.] The valves covered by this LCO are listed along with their associated stroke times in the FSAR (Ref. 4).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, check valves have flow through the valve secured, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 5.

Purge valves with resilient seals must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designated safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for [48] inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow paths containing these valves may not be opened under administrative controls. A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.

BASES

ACTIONS (continued)

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable , since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, [except for purge valve leakage not within limit], the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration 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 the 4 hour Completion Time. The specified time period is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device

BASES

ACTIONS (continued)

manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating this Condition is only applicable to those penetration flow paths with two [or more] containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows the devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

<u>B.1</u>

With two [or more] containment isolation valves in one or more penetration flow paths inoperable, [except for purge valve leakage not within limit], the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in

BASES

ACTIONS (continued)

accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two [or more] containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. 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 supporting containment OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference 6. This Note is necessary since this Condition

BASES

ACTIONS (continued)

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 by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once verified to be in the proper position, is small.

[D.1, D.2, and D.3

In the event one or more containment purge valves in one or more penetration flow paths are not within the purge valve leakage limits, purge valve leakage must be restored to within limits or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a [closed and de-activated automatic valve, closed manual valve, and blind flange]. A purge valve with resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.6. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and potentially capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered

BASES

ACTIONS (continued)

reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the containment purge valve with resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.3.6 must be performed at least once every [ ] days. This provides assurance that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.6, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 8). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per [ ] days was chosen and has been shown acceptable based on operating experience.]

Required Action D.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

<u>E.1 and E.2</u>

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

[<u>SR 3.6.3.1</u>

Each [48] inch containment purge valve is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of the purge

BASES

SURVEILLANCE REQUIREMENTS (continued)

valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A containment purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 7), related to containment purge valve use during unit operations. In the event purge valve leakage requires entry into Condition D, the Surveillance permits opening one purge valve in a penetration flow path to perform repairs.]

SR 3.6.3.2

This SR ensures that the minipurge 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 minipurge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The minipurge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside 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 containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves open under administrative

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is low.

SR 3.6.3.4

This SR requires verification that 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. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves 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.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate

BASES

SURVEILLANCE REQUIREMENTS (continued)

OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. [The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program or 92 days.]

SR 3.6.3.6

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option [A][B] is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of once per 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 8).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (greater than that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The [18] 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 [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.8

- REVIEWER'S NOTE -

This SR is only required for those units with resilient seal purge valves allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices on the valves that are not permanently installed.

Verifying that each [48] inch containment purge valve is blocked to restrict opening to ≤ [50%] is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 3 and 4. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of [recently] irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The [18] month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

REFERENCES

1. 10 CFR 20.
2. FSAR, Section [5.6].
3. FSAR, Sections [14.1 and 14.2].
4. FSAR, Section [5.3].
5. FSAR, Section [5.3].
6. Standard Review Plan 6.2.4
7. Generic Issue B-24.
8. Generic Issue B-20.

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 inadvertent actuation of the Containment Spray System. Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.
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. The worst-case LOCA generates larger mass and energy release than the worst-case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1). The initial pressure condition used in the containment analysis was [17.7] psia ([3.0] psig). This resulted in a maximum peak pressure from a LOCA of [53.9] psig. The LCO limit of [3.0] psig ensures that, in the event of an accident, the design pressure of [55] psig for containment is not exceeded. In addition, the building was designed for an internal pressure equal to [3] psig above external pressure during a tornado. The containment was also designed for an internal pressure equal to [2.5] psig below external pressure, to withstand the resultant pressure drop from an accidental actuation of the Containment Spray System. The LCO limit of [-2.0] psig ensures that operation within the design limit of [-2.5] psig is maintained (Ref. 2). For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling Systems during the core reflood phase of a LOCA analysis increases with increasing

BASES

APPLICABLE SAFETY ANALYSES (continued)

containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Maintaining containment pressure 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 greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within design basis limits is essential to ensure initial conditions assumed in the accident analysis 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 MODES 5 and 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits 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.

B.1 and B.2

If containment pressure cannot be restored 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

BASES

ACTIONS (continued)

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 operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. FSAR, Section [14.2].
2. 10 CFR 50, Appendix K.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND	The containment structure serves to contain radioactive material, which 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 analysis. This LCO ensures that initial conditions assumed in the analysis of a DBA are not violated during unit operations. The total amount of energy to be removed from the 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 higher the resultant 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. Average air temperature is also used to establish the containment environmental qualification operating envelope. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analysis for containment. Several accidents (primarily LOCA and SLB) result in a marked increase in containment temperature and pressure due to energy release within the containment. Of these, the LOCA results in the greatest sustained increase in containment temperature. By maintaining containment air temperature at less than the initial temperature assumed in the LOCA analysis, the reactor building design condition will not be exceeded. The LOCA that was identified as presenting the greatest challenge to containment OPERABILITY was a cold leg Reactor Coolant System break, of specified size, at a reactor coolant pump suction.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 maintained below the containment design temperature. 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.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment

BASES

SURVEILLANCE REQUIREMENTS (continued)

average air temperature, an arithmetic average is calculated, using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray and Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1), or other documents that were appropriate at the time of licensing (identified on a unit specific basis).

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and Containment Cooling System provide redundant containment heat removal operation. The Containment Spray System and Containment Cooling System provide redundant methods to limit and maintain post accident conditions to less than the containment design values.

<u>Containment Spray System</u>

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design basis. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The borated water storage tank (BWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray System pump suction is manually transferred from the BWST to the containment sump.

The Containment Spray System provides a spray of relatively cold borated water mixed with sodium hydroxide from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature and to reduce the concentration of fission products in

BASES

BACKGROUND (continued)

the containment atmosphere during a DBA. In the recirculation mode of operation, heat is removed from the containment sump water by the decay heat removal coolers. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment heat removal.

The Containment Spray System is actuated automatically by a containment High-High pressure signal coincident with a containment high pressure signal and a low pressure injection signal. An automatic actuation opens the Containment Spray System pump discharge valves and starts the two Containment Spray System pumps. [A manual actuation of the Containment Spray System requires the operator to actuate two separate switches on the main control board to begin the same sequence.]

Containment Cooling System

The Containment Cooling System consists of three containment cooling trains connected to a common duct suction header with four vertical return air ducts. Each cooling train is equipped with demisters, cooling coils, and an axial flow fan driven by a two speed water cooled electric motor. Each unit connection (two per unit) to the common header is provided with a backpressure damper for isolation purposes.

During normal operation, two containment cooling trains are required to operate. The third unit is on standby and isolated from the operating units by means of the backpressure dampers. The swing unit is equipped with a transfer switch. It can be manually placed to either the "A" or "B" power train to operate in case one of the operating units fails. Upon receipt of an emergency signal, the two operating cooling fans running at high speed will automatically stop. The two cooling unit fans connected to the ESF buses will automatically restart and run at low speed, provided normal or emergency power is available.

In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically in slow speed if they are not already running. If they are running at high (normal) speed, the fans automatically stop and restart in slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher density atmosphere.

BASES

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit 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. The postulated DBAs are analyzed, with regard to containment ESF systems, assuming the loss of one ESF bus. This is the worst-case single active failure, resulting in one train of the Containment Spray System and one train of the Containment Cooling System being inoperable.

The analysis and evaluation show that, under the worst-case scenario, the highest peak containment pressure is [53.9] psig (experienced during a LOCA). The analysis shows that the peak containment temperature is [276]°F (experienced during a LOCA). Both results are less than the design values. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of [2568] MWt, one containment spray train and one containment cooling train operating, and initial (pre-accident) conditions of [130]°F and [17.7] psia. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a [2.5] psig containment pressure drop and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analyses is based on a response time associated with exceeding the containment pressure High-High setpoint coincident with a high pressure injection signal to achieve full flow through the containment spray nozzles. The Containment Spray System total response time of [56] seconds includes diesel generator (DG) startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 2).

Containment cooling train performance for post accident conditions is given in Reference 3. The result of the analysis is that each train can provide 33% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 4.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The modeled Containment Cooling System actuation from the containment analysis is based on a response time associated with exceeding the containment pressure high setpoint to achieve full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time of [25] seconds includes signal delay, DG startup (for loss of offsite power), and service water pump startup times (Ref. 3).

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, a minimum of one containment cooling train and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits. Additionally, one containment spray train is required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs.

Each Containment Spray System typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered Safety Features Actuation System signal and manually transferring suction to the containment sump.

Each Containment Cooling System typically includes demisters, cooling coils, dampers, an axial flow fan driven by a two speed water cooled electrical motor, instruments, and controls to ensure an OPERABLE 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 containment spray trains and containment cooling trains.

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, the Containment Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

BASES

ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and the low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, Completion Times, for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 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 to attempt restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one of the required containment cooling trains inoperable, the inoperable containment cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

BASES

ACTIONS (continued)

The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

D.1

With two of the required containment cooling trains inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. The components in this degraded condition (both spray trains are OPERABLE or else Condition F is entered) provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

E.1 and E.2

If the Required Actions and associated Completion Times of Condition C or D of this LCO 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.

F.1

With two containment spray trains or any combination of three or more containment spray and containment cooling trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray 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 position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This 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 potentially being mispositioned are in the correct position.

SR 3.6.6.2

Operating each [required] containment cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of a significant degradation of the containment cooling trains occurring between surveillances and has been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying that each [required] containment cooling train provides an essential raw water cooling flow rate of ≥ [1780] gpm to each cooling unit provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 1). The Frequency was developed considering the known reliability of the Cooling Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5). Since the Containment Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test

BASES

SURVEILLANCE REQUIREMENTS (continued)

confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The [18] month Frequency is based on the need to perform these Surveillances 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. Operating experience has shown that these components usually pass the Surveillances when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.7

This SR requires verification that each [required] containment cooling train actuates upon receipt of an actual or simulated actuation signal. The [18] month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the [18] month Frequency.

SR 3.6.6.8

With the containment spray header isolated and drained of any solution, low pressure air or smoke can be blown through test connections. Performance of this Surveillance demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzles, a test at [the first refueling and at] 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. FSAR, Section [14.1].
3. FSAR, Section [6.3].
4. FSAR, Section [14.2].
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA).

The Containment Spray System and Spray Additive System perform no function during normal operations. In the event of an accident such as a loss of coolant accident (LOCA), however, the Spray Additive System will be automatically actuated upon a high containment pressure signal by the Engineered Safety Features Actuation System.

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Sodium hydroxide (NaOH), because of its stability when exposed to radiation and elevated temperature, is the preferred spray additive.

The spray additive tank is designed and located to permit gravity draining into the Containment Spray System. Both Containment Spray System pumps initially take suction from the borated water storage tank (BWST) via two independent flow paths. The spray additive tank has a common header that splits and feeds each of the Containment Spray System suction lines. The system is designed to inject at a rate commensurate with the draining rate of the BWST so that all borated water injected is mixed with NaOH.

The flow rate is proportioned to provide a spray solution with a pH between [7.2 and 11.0] (Ref. 1). This range of alkalinity was established not only to aid in removal of airborne iodine, but also to minimize the corrosion of mechanical system components that would occur if the acidic borated water were not buffered. The pH range also considers the environmental qualification of equipment in containment that may be subjected to the spray.

BASES

APPLICABLE SAFETY ANALYSES

The containment Spray Additive System is essential to the effective removal of airborne iodine within containment following a DBA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value following the accident. The analysis assumes that most of the containment volume is covered by the spray.

The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.6, "Containment Spray and Cooling Systems."

The DBA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable and that the entire spray additive tank volume is added to the remaining Containment Spray System flow path.

In the evaluation of the worst-case LOCA, the safety analysis assumed that an alkaline containment spray effectively reduced the airborne iodine.

Each Containment Spray System suction line is equipped with its own gravity feed from the spray additive tank. Therefore, in the event of a single failure within the Spray Additive System (i.e., suction valve failure), NaOH will still be mixed with the borated water, establishing the alkalinity essential to effective iodine removal.

The Spray Additive System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow until the Containment Spray System suction path is switched from the BWST to the containment sump and to raise the average spray solution pH to a level conducive to iodine removal. The average spray solution pH is between [7.2 and 11.0]. This pH range maximizes the effectiveness of the iodine removal mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to 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 in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With the containment Spray Additive System inoperable, the system must be restored to OPERABLE status within 72 hours. The pH adjustment of the Containment Spray System for corrosion protection and iodine removal enhancement is reduced in this Condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst-case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System 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 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 for restoration of the Spray Additive System 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 3.6.7.1

Verifying the correct alignment of spray additive manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment capable of potentially being mispositioned are in the correct position.

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the BWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency is based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, such that there is a high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The concentration of NaOH in the spray additive tank must be determined by chemical analysis. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. The [18] 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 the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.5

To ensure that the correct pH level is established in the borated water solution provided by the Containment Spray System, the flow [rate] in the Spray Additive System is verified once per 5 years. This SR provides assurance that the correct amount of NaOH will be metered into the flow path upon Containment Spray System initiation. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow [rate].

REFERENCES

1. FSAR, Section [6.2].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Recombiners

BASES

BACKGROUND

Permanently installed hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor is returned to the containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammability limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiners are provided. Each consists of controls located in the control room, a power supply, and a recombiner located external to containment. The recombiners have no moving parts. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. Air flows through the unit at approximately 3000 cfm at a maximum supply temperature of 120°F. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.1 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than a concentration of 4.1 v/o following a DBA. This control would prevent a hydrogen burn inside containment, thus ensuring the pressure and temperature assumed in the accident analysis are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate within containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant,

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump,

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space), or

d. Corrosion of metals exposed to Containment Spray System and Emergency Core Cooling Systems solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 1 are used to maximize the amount of hydrogen calculated. These evaluations demonstrate approximately 10 days are needed for hydrogen concentration to increase to 4.1 v/o post LOCA.

The hydrogen recombiners satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst-case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst-case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations. Therefore, hydrogen recombiners are not required in these MODES.

BASES

ACTIONS

A.1

With one hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in a reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note stating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one hydrogen recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

B.1 and B.2

- REVIEWER'S NOTE -

This Condition is only allowed for units with an alternate hydrogen control system acceptable to the technical staff.

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by [the containment Hydrogen Purge System/hydrogen recombiner/Hydrogen Ignitor System/Hydrogen Mixing System/Containment Air Dilution System/Containment Inerting System]. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist.

BASES

ACTIONS (continued)

- REVIEWER'S NOTE -

The following is to be used if a non-Technical Specification alternate hydrogen control function is used to justify this Condition: In addition, the alternate hydrogen control system capability must be verified every 12 hours thereafter to ensure its continued availability.

[Both] the [initial] verification [and all subsequent verifications] may be performed as an administrative check, by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

C.1

If the inoperable hydrogen recombiner(s) 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. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Performance of a system functional test for each hydrogen recombiner ensures that the recombiners are operational and can obtain and sustain the temperature necessary for hydrogen recombination. In particular, this SR requires verification that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum for approximately 2 minutes and power verified to be ≥ 60 kW. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.8.2

This SR ensures that there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The [18] month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.8.3

This SR requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms. The [18] month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. Regulatory Guide 1.7, Revision [1].

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 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.

Nine MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section [5.2] (Ref. 1). The MSSV rated capacity passes the full steam flow at 112% RTP with the valves full open. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-1 in the accompanying LCO, so that only the needed number of valves will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open all valves following a turbine reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis of the MSSVs comes from Reference 2 and its purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in the FSAR, Section [15.2] (Ref. 3). Of these, the full power turbine trip coincident with a loss of condenser heat sink is the limiting AOO. For this event, the Condenser Circulating Water System is lost and, therefore, the Turbine Bypass Valves are not available to relieve Main Steam System pressure. Similarly, MSSV relief capacity is utilized in the FSAR for mitigation of the following events:

a. Loss of main feedwater,

b. Steam line break,

c. Steam generator tube rupture, and

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. Excessive heat removal due to feedwater system malfunction.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The MSSVs setpoints are established to prevent overpressurization as discussed in the Applicable Safety Analysis section of these Bases. The LCO requires all MSSVs to be OPERABLE to ensure compliance with the ASME Code following DBAs initiated at full power. Operation with less than a full complement of MSSVs requires limitations on unit THERMAL POWER and adjustment of the Reactor Protection System (RPS) trip setpoints. This effectively limits the Main Steam System steam flow while the MSSV relieving capacity is reduced due to valve inoperability. To be OPERABLE, lift setpoints must remain within limits, according to Table 3.7.1-1 in the accompanying LCO.

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, according to Table 3.7.1-1 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 the design safety function to mitigate the consequences of accidents that could result in a challenge to the RCPB.

APPLICABILITY

In MODE 1 above [18]% RTP, the number of MSSVs per steam generator required to be OPERABLE must be within the acceptable region, according to Figure 3.7.1-1 in the accompanying LCO. Below [18]% RTP in MODES 1, 2, and 3, only two MSSVs are required OPERABLE per steam generator.

In MODES 4 and 5, there is no credible transient 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.

BASES

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets ASME Code requirements for the power level. Operation may continue, provided the ALLOWABLE THERMAL POWER and RPS nuclear overpower trip setpoint are reduced by the application of the following formulas:

$$RP = [\ Y / Z] \times 100\%$$

and

$$SP = [\ Y / Z\ ] \times W$$

where:

W = Nuclear overpower trip setpoint for four pump operation as specified in LCO 3.3.1, "Reactor Protection System (RPS),"

Y = Total OPERABLE MSSV relieving capacity per steam generator based on a summation of individual OPERABLE MSSV relief capacities per steam generator [lb/hour];

Z = Required relieving capacity per steam generator of [6,585,600] lb/hour,

RP = Reduced power requirement (not to exceed RTP), and

SP = Nuclear overpower trip setpoint (not to exceed W).

These equations are graphically represented in Figure 3.7.1-1, in the accompanying LCO. Operation is restricted to the area below and to the right of line BCDE.

The operator should limit the maximum steady state power level to some value slightly below this setpoint to avoid an inadvertent overpower trip.

The 4 hour Completion Time for Required Action A.1 is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 32 hours is allowed in Required Action A.2 to

BASES

ACTIONS (continued)

reduce the setpoints. The Completion Time of 36 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

B.1 and B.2

With one or more MSSVs inoperable, a verification by administrative means that at least [two] required MSSVs per steam generator are OPERABLE, with each valve from a different lift setting range, is performed.

If the MSSVs cannot be restored to OPERABLE status in 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 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4) requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required for MSSVs:

a. Visual examination,

b. Seat tightness determination,

c. Setpoint pressure determination (lift setting),

d. Compliance with owner's seat tightness criteria, and

e. Verification of the balancing device integrity device on balanced valves.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The ANSI/ASME Standard requires the testing of all valves every 5 years, with a minimum of 20% of the valves tested every 24 months. Reference 4 provides the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-1 allows a ± [3]% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

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. FSAR, Section [5.2].
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. FSAR, Section [15.2].
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. ANSI/ASME OM-1-1987.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside of, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Turbine Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a Steam and Feedwater Rupture Control System signal generated by either low steam generator pressure or steam generator to feedwater differential pressure. The MSIVs fail closed on loss of control or actuation power. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section [10.3] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, as discussed in the FSAR, Section [6.2] (Ref. 2). It is also influenced by the accident analysis of the SLB events presented in the FSAR, Section [15.4] (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (i.e., the failure of one MSIV to close on demand).

The limiting case for the containment analysis is the SLB inside containment with a loss of offsite power following turbine trip and failure of the MSIV on the affected steam generator to close. At 100% RTP, the steam generator inventory and temperature are at their maximum, maximizing the mass and energy release to the containment.

Due to reverse flow, failure of the MSIV to close contributes to the total release of the additional mass and energy in the steam headers downstream of the other MSIV. Other failures considered are the failure of a main feedwater isolation valve to close, and failure of an emergency diesel generator (EDG) to start.

APPLICABLE SAFETY ANALYSES (continued)

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 full power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed. Significant single failures considered include failure of an MSIV to close, failure of an EDG, and failure of an HPI pump.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An HELB, an SLB, or main feedwater line breaks (FWLBs), inside containment. In order to maximize the mass and energy release into the containment, the analysis assumes the MSIV in the affected steam generator remains open. For this scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIV in the intact loop.

b. An SLB outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the steam bypass valves will also terminate on closing the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generator. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the steam

BASES

APPLICABLE SAFETY ANALYSES (continued)

generator with the ruptured tube below the MSIVs' setpoints, a necessary step toward isolating flow through the rupture.

e. The MSIVs are also utilized during other events such as an FWLB.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the MSIV in both steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 4).

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1 and in MODES 2 and 3 with any MSIVs open, when there is significant mass and energy in the RCS and steam generator; therefore, the MSIVs must be OPERABLE or closed. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low. Therefore, the MSIVs are not required to be OPERABLE.

In MODES 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore the component to OPERABLE status within [8] hours. Some repairs can be made to the MSIV with the unit hot. The [8] hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the MSIVs occurring during this time interval. The turbine stop valves are available to provide the required isolation for the postulated accidents.

The [8] hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other

BASES

ACTIONS (continued)

containment isolation valves in that the closed system provides an additional means for containment isolation.

B.1

If the MSIV cannot be restored to OPERABLE status within [8] hours, the unit must be placed in MODE 2 and the inoperable MSIV closed within the next 6 hours. The Completion Times are reasonable, based on operating experience, to reach MODE 2.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The [8] hour Completion Time is consistent with that allowed in Condition A.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the MSIV cannot be restored to OPERABLE status or closed in 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 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time of each MSIV is ≤ [6] seconds. 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, because the MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. As the MSIVs are not to be tested at power, they are exempt from the ASME Code, Section XI (Ref. 5) requirements during operation in MODES 1 and 2.

The Frequency for this SR 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 delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MSIV testing is every [18] months. The [18] month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [10.3].
2. FSAR, Section [6.2].
3. FSAR, Section [15.4].
4. 10 CFR 100.11.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.3 [Main Feedwater Stop Valves (MFSVs), Main Feedwater Control Valves (MFCVs), and Associated Startup Feedwater Control Valves (SFCVs)]

BASES

BACKGROUND

The main feedwater isolation valves (MFIVs) for each steam generator consist of the MFSVs, MFCVs, and the SFCVs. The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Closure of the MFIVs terminates flow to both steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the feedwater lines downstream of the MFIVs will be mitigated by their closure. Closing the MFIVs and associated bypass valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment and reducing the cooldown effects for SLBs.

The MFIVs close on receipt of a Steam and Feedwater Rupture Control System (SFRCS) signal generated by either low steam generator pressure or steam generator/feedwater differential pressure. The MFIVs can also be closed manually.

[The MFIVs and associated bypass valves close on receipt for a safety injection - low T_{avg} coincident with reactor trip or steam generator water level - high high signal. They may also be actuated manually. In addition to the MFIVs and associated bypass valves, a check valve inside containment is available to isolate the feedwater line penetrating containment and to ensure that the consequences of events do not exceed the capacity of the containment heat removal systems.]

A description of the MFIVs is found in the FSAR, Section [10.4.7] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs is established by the analysis for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs may also be relied on to terminate a steam break for core response analysis and excess feedwater event upon the receipt of a steam generator water level - high signal.

Failure of an MFIV to close following an SLB, FWLB, or excess feedwater event, can result in additional mass and energy being delivered to the

BASES

APPLICABLE SAFETY ANALYSES (continued)

steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures that the MFIVs will isolate MFW flow to the steam generators following a FWLB or a main steam line break. These valves will also isolate the nonsafety related portions from the safety related portions of the system.

[Two] [MFSVs], [MFCVs], [or associated SFCVs] are required to be OPERABLE. The MFIVs are considered OPERABLE when the 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. If the SFRCS on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

The [MFSVs], [MFCVs], [or associated SFCVs] must be OPERABLE whenever there is significant mass and energy in the RCS and steam generators. This ensures that in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator.

In MODES 1, 2, and 3, the [MFSVs], [MFCVs], [or associated SFCVs] are required to be OPERABLE in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed, they are already performing their safety function.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the [MFSVs], [MFCVs], [or associated SFCVs] are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one [MFSV] in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or

BASES

ACTIONS (continued)

isolate inoperable affected valves within [8 or 72] hours. When these valves are closed or isolated, they are performing their required safety function.

[For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable to close the MFIV or its associated bypass valve which includes performing a controlled unit shutdown to MODE 2. The Completion Time is reasonable, based on operating experience, to reach MODE 2 from full power conditions with the MFIVs closed, in an orderly manner and without challenging unit systems.]

The [72] hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The [72] hour Completion Time is reasonable, based on operating experience.

Inoperable [MFSVs] that are closed or isolated, must be verified on a periodic basis that they are closed or 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 valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one [MFCV] in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within [8 or 72] hours. When these valves are closed or isolated, they are performing their required safety function.

[For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable, based on operating experience, to close the MFIV or its associated bypass valve.]

The [72] hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable [MFCVs] that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to

BASES

ACTIONS (continued)

ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

C.1 and C.2

With one [SFCV] in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within [8 or 72] hours. When these valves are closed or isolated, they are performing their required safety function.

[For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable, based on operating experience, to close the MFIV or its associated bypass valve.]

The [72] hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable SFCVs that are closed or isolated must be verified on a periodic basis that they are closed or 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 valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

D.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. Although the containment can be isolated with the failure to two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path and as such is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. The 8 hour Completion Time is reasonable, based on operating experience, to close the MFIV or otherwise isolate the affected flow path.

BASES

ACTIONS (continued)

E.1 and E.2

If the [MFSVs], [MFCVs], and [associated SFCVs] cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time, the unit must be in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each [MFSV], [MFCV], and [associated SFCV] is ≤ 7 seconds.

The [MFSV], [MFCV], and [associated SFCV] 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 [MFSV], [MFCV], and [associated SFCV] should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR.

The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 3.7.3.2

This SR verifies that each [MFSV, MFCV, and associated SFCV] can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Frequency for this SR is every [18] months. The [18] month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, this Frequency is acceptable from a reliability statndpoint.

BASES

REFERENCES	1. FSAR, Section [10.4.7].
	2. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Vent Valves (AVVs)

BASES

BACKGROUND

The AVVs provide a method for cooling the unit to decay heat removal (DHR) entry conditions, should the preferred heat sink via the Turbine Bypass System to the condenser not be available, as discussed in the FSAR, Section [10.3] (Ref. 1). This is done in conjunction with the Emergency Feedwater System, providing cooling water from the condensate storage tank (CST). The AVVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Turbine Bypass System.

[The AVVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation.]

The AVVs are equipped with pneumatic controllers to permit control of the cooldown rate.

[The AVVs are provided with a pressurized gas supply of bottled nitrogen that, on loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the AVVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the AVVs for the time required for Reactor Coolant System (RCS) cooldown to DHR entry conditions.]

A description of the AVVs is found in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the AVVs is established by the capability to cool the unit to MODE 3. The design rate of [75]°F per hour is applicable for both steam generators, each with one AVV. This rate is adequate to cool the unit to DHR entry conditions with only one AVV and one steam generator utilizing the cooling water supply available in the CST.

In the accident analysis presented in Reference 1, the AVVs are assumed to be used by the operator to cool down the unit to MODE 3 for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the AVVs and the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator's pressure and temperature below the design value. This is about 30 minutes following initiation of an event; however, this may be less for a steam generator tube rupture (SGTR)

BASES

APPLICABLE SAFETY ANALYSES (continued)

event. Some initiating events falling into this category are a main steam line break upstream of the main steam isolation valves, a feedwater line break, and an SGTR event (although the AVVs on the affected steam generator may still be available following an SGTR event).

For the recovery from an SGTR event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to DHR conditions for this event, and also for other accidents. Thus, the SGTR is the limiting event for the AVVs. The number of AVVs required to be OPERABLE to satisfy the SGTR accident analysis requirements depends upon the consideration of any single failure assumptions regarding the failure of one AVV to open on demand.

[The design must accommodate the single failure of one AVV to open on demand, thus each steam generator must have at least one AVV. The AVVs are equipped with manual block valves in the event an AVV spuriously fails open, or fails to close during use.]

The AVVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

[Two] AVVs [lines per steam generator] are required to be OPERABLE. Failure to meet the LCO can result in the inability to cool the unit to DHR entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An AVV is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and is capable of fully opening and closing on demand.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the AVVs are required to be OPERABLE.

In MODES 5 and 6, an SGTR is not a credible event.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

With one AVV [line] inoperable, action must be taken to restore the inoperable AVV to OPERABLE status. The 7 day Completion Time allows for redundant capability afforded by the remaining OPERABLE AVV and a nonsafety grade backup in the Steam Bypass System and MSSVs.

[B.1

With more than one AVV [line] inoperable, action must be taken to restore [all but one] AVV [lines] to OPERABLE status. As the block valve can be closed to isolate an AVV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable AVV [lines], based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the AVV [lines].]

C.1 and C.2

If the AVV [lines] cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within [24] hours, without reliance upon the steam generator for heat removal. 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

To perform a controlled cooldown of the RCS, the AVVs must be able to be opened either remotely or locally and throttled through their full range. This SR ensures that the AVVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an AVV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.7.4.2

The function of the block valve is to isolate a failed open AVV. Cycling the block valve closed and open demonstrates its ability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.]

REFERENCES

1. FSAR, Section [10.3].

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The EFW pumps take suction through separate and independent suction lines from the condensate storage tank (CST) (LCO 3.7.6, "Condensate Storage Tank (CST)"), and pump to the steam generator secondary side through the EFW nozzles. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)"), or atmospheric vent valves (AVVs) (LCO 3.7.4, "Atmospheric Vent Valves (AVVs)"). If the main condenser is available, steam may be released via the Turbine Bypass System and recirculated to the CST.

[The following system description is provided as an example. Actual system description should be provided by the specific unit. The EFW System consists of two turbine driven EFW pumps, each of which provides a nominal 100% capacity, and one nonsafety grade motor driven EFW pump. The steam turbine driven EFW pumps receive steam from either of the two main steam headers, upstream of the main steam isolation valves (MSIVs). The EFW System supplies a common header capable of feeding either or both steam generators. The 100% capacity is sufficient to remove decay heat and cool the unit to decay heat removal (DHR) entry conditions. The EFW System normally receives a supply of water from the CST. A safety grade source of water is also supplied by the Service Water System (SWS). Automatic valves on the supply piping open on low pressure in the supply piping to transfer the water supply from the CST to the SWS. A third source of water can be supplied by manually aligning the fire protection header to the EFW pump suction.] Thus, the requirements for diversity in motive power sources for the EFW System are met.

The EFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The EFW System is designed to supply sufficient water to cool the unit to DHR entry conditions with steam being released through the ADVs or condenser.

BASES

BACKGROUND (continued)

The EFW actuates automatically on low steam generator level, low steam generator pressure, or loss of four reactor coolant pumps.

The EFW System is discussed in the FSAR, Sections [9.2.7] and [9.2.8] (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The EFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the EFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3%.

In addition, the EFW System must supply enough makeup water to replace steam generator secondary inventory being lost as steam as the unit cools to MODE 4 conditions. Sufficient EFW flow must also be available to account for flow losses such as pump recirculation and line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the EFW System are as follows:

a. Feedwater line break (FWLB) and

b. Loss of main feedwater.

In addition, the minimum available EFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident.

[The EFW System design is such that it can perform its function following a loss of the turbine driven main feedwater pumps or an FWLB, combined with a loss of normal or reserve electric power.]

The EFW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO provides assurance that the EFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. [Three] independent EFW pumps, in two diverse trains are

BASES

LCO (continued)

required to be OPERABLE to ensure the availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. [This is accomplished by powering two pumps by steam driven turbines supplied with steam from a source not isolated by the closure of the MSIVs, and one pump from a power source that, in the event of loss of offsite power, is supplied by the emergency diesel generator.]

The EFW System is considered to be OPERABLE when the components and flow paths required to provide EFW flow to the steam generators are OPERABLE. This requires that the [two] turbine driven EFW pump(s) be OPERABLE with redundant steam supplies from each of the main steam lines upstream of the MSIVs and capable of supplying EFW flow to either of the two steam generators. The [nonsafety grade] motor driven EFW pump(s) and [the] associated flow path(s) to the EFW System [are] also required to be OPERABLE. The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE. The primary and secondary sources of water to the EFW System are required to be OPERABLE. The associated flow paths from the EFW System primary and secondary sources of water to all EFW pumps also are required to be OPERABLE.

The LCO is modified by a Note indicating that one EFW train, which includes a motor driven EFW pump, is required in MODE 4. This is because of reduced heat removal requirement, the short duration of MODE 4 in which feedwater is required, and the insufficient steam supply available in MODE 4 to power the turbine driven EFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the EFW System is required to be OPERABLE and to function in the event that the main feedwater is lost. In addition, the EFW System is required to supply enough makeup water to replace the steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

In MODE 4, with RCS temperature above [212]°F, the EFW System may be used for heat removal via the steam generators. In MODE 4, the steam generators are used for heat removal until the DHR System is in operation.

In MODES 5 and 6, the steam generators are not used for DHR and the EFW System is not required.

BASES

ACTIONS

[A.1

With one of the two steam supplies to the turbine driven EFW pump inoperable, or if a turbine driven pump is inoperable while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. For the inoperability of a steam supply to the turbine driven EFW pump, the 7 day Completion time is reasonable since there is a redundant steam supply line for the turbine driven pump.

b. For the inoperability of a turbine driven EFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion time is reasonable due to the minimal decay heat levels in this situation.

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven EFW pump while in MODE 3 immediately following refueling, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven EFW pumps, and due to the low probability of an event requiring the use of the turbine driven EFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

Condition A is modified by a Note which limits the applicability of the Condition to when the unit has not entered MODE 2 following a refueling. Condition A allows one EFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.]

B.1

When one of the required EFW trains (pump or flow path) is inoperable, action must be taken to restore the train to OPERABLE status within

BASES

ACTIONS (continued)

72 hours. This Condition includes the loss of two steam supply lines to one of the turbine driven EFW pumps. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of a DBA occurring during this time period. The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The <u>AND</u> connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

<u>C.1 and C.2</u>

When either Required Action A.1 or Required Action B.1 cannot be completed within the required Completion Time, [or when two EFW trains are inoperable in MODE 1, 2, or 3,] the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within [18] 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.

In MODE 4, with two EFW trains inoperable, operation is allowed to continue because only one motor driven EFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate DHR.

<u>D.1</u>

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until at least one EFW train is restored to OPERABLE status.

With [all] EFW trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with

BASES

ACTIONS (continued)

nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the units into a less safe condition.

E.1

In MODE 4, either the steam generator loops or the DHR loops can be used to provide heat removal, which is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With one EFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by Section XI of the ASME Code (Ref. 3). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test is performed on recirculation flow.

This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component

BASES

SURVEILLANCE REQUIREMENTS (continued)

OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing in the ASME Code, Section XI (Ref. 3), at 3 month intervals, satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR verifies that EFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates a Steam and Feedwater Rupture Control System (SFRCS) signal by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The [18] month Frequency is also acceptable based on operating experience and design reliability of the equipment. This SR is modified by a Note that states the SR is not required to be met in MODE 4. In MODE 4, the required AFW train is already aligned and operating. This SR is modified by [a] [two] Note[s]. [Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.] [The] Note [2] states that the SR is not required to be met in MODE 4. [In MODE 4, the required pump is already operating and the autostart function is not required.] [In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.]

SR 3.7.5.4

This SR verifies that the turbine driven EFW pumps start in the event of any accident or transient that generates an SFRCS signal by demonstrating that each turbine driven EFW pump starts automatically on an actual or simulated actuation signal. These pumps are not required in MODE 4. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance

BASES

SURVEILLANCE REQUIREMENTS (continued)

were performed with the reactor at power. This SR is modified by [a] [two] Note[s]. [Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.] [The] Note [2] states that the SR is not required to be met in MODE 4. [In MODE 4, the required pump is already operating and the autostart function is not required.] [In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.]

- REVIEWER'S NOTE -

Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.

SR 3.7.5.5

This SR ensures that the EFW System is properly aligned by verifying the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in any combination of MODES 5 or 6, or defueled. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, in view of other administrative controls to ensure that the flow paths are OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generator is properly aligned. (This SR is not required by those units that use EFW for normal startup and shutdown.)

[SR 3.7.5.6 and SR 3.7.5.7

For this facility, the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the EFW pump suction pressure interlocks are as follows:

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and

BASES

SURVEILLANCE REQUIREMENTS (continued)

non-Technical Specifications tests at least once per refueling interval with applicable extensions.]

REFERENCES

1. FSAR, Section [9.2.7].
2. FSAR, Section [9.2.8].
3. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Emergency Feedwater (EFW) System (LCO 3.7.5, "Emergency Feedwater (EFW) System"). The steam produced is released to the atmosphere by the main steam safety valves (MSSVs) or the atmospheric vent valves.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge to the condenser by the nonsafety grade path of the turbine bypass valves. The condensed steam is returned to the CST by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.

Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena, as well as missiles that might be generated by natural phenomena. The CST is designed to Seismic Category I to ensure availability of the feedwater supply. Feedwater is also available from an alternate source(s).

A description of the CST is found in the FSAR, Section [9.2.60] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and cool down the unit following all events in the accident analysis, as discussed in the FSAR, Chapters [6] and [15] (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs, followed by a cooldown to decay heat removal (DHR) entry conditions at the design cooldown rate.

The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite power. Single failures that also affect this event include the following:

a. Failure of the diesel generator powering the motor driven EFW pump to the unaffected steam generator (requiring additional steam to drive the remaining EFW pump turbine) and

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. Failure of the steam driven EFW pump (requiring a longer time for cooldown using only one motor driven EFW pump).

These are not usually the limiting failures in terms of consequences for these events.

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

To satisfy accident analysis assumptions, the [two] CSTs must contain sufficient cooling water to remove decay heat for 13 hours following a reactor trip from 102% RTP and then to cool down the RCS to DHR System entry conditions, assuming a coincident loss of offsite power and most adverse single failure. While so doing, the CSTs must retain sufficient water to ensure adequate net positive suction head for the EFW pump(s) during the cooldown, to account for any losses from the steam driven EFW pump turbine, as well as losses incurred before isolating EFW to a broken line.

The level required is equivalent to a usable volume of [250,000] gallons, which is based on holding the unit in MODE 3 for 13 hours, followed by a cooldown to DHR System entry conditions.

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

APPLICABILITY

In MODES 1, 2, 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODES 5 and 6, the CST is not required because the EFW System is not required.

ACTIONS

A.1 and A.2

As an alternative to unit shutdown, the OPERABILITY of the backup water supply should be verified within 4 hours and once every 12 hours thereafter. The OPERABILITY of the backup feedwater supply must include verification, by administrative means, of the OPERABILITY of flow paths from the backup supply to the EFW pumps and availability of the required volume of water in the backup supply. The CST must be restored to OPERABLE status within 7 days because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to

BASES

ACTIONS (continued)

verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period, requiring the use of the water from the CST(s).

B.1 and B.2

If the CST cannot be restored to OPERABLE status in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply, with the DHR System in operation. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generators for heat removal, within [24] hours. This allows an additional 6 hours for the DHR System to be placed in service after entering MODE 4.

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 3.7.6.1

This SR verifies that the CST(s) contains the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in CST levels.

REFERENCES

1. FSAR, Section [9.2.6].
2. FSAR, Chapter [6].
3. FSAR, Chapter [15].

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System, and thus to the environment.

A typical CCW System is arranged as two independent full capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. A surge tank in the system provides sufficient net positive suction head for each pump and isolation of nonessential components on a low tank level signal. The pump in each train is automatically started on receipt of a safety feature actuation signal, and all nonessential components are isolated.

Additional information on the design and operation of the CCW System, along with a list of the components served, is presented in the FSAR, Section [9.2.2] (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the reactor via the [decay heat removal (DHR) heat exchanger]. This may utilize the DHR System during a normal or post accident cooldown and shutdown, or during the recirculation phase following a loss of coolant accident.

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is to provide cooling water to the Emergency Core Cooling System and emergency diesel generators (EDGs) during DBA conditions. The CCW System also supplies cooling water to EDGs during a loss of offsite power.

The CCW System is designed to perform its function with a single failure of any active component assuming a loss of offsite power.

The CCW System also functions to cool the unit from [DHR] entry conditions (T_{cold} < [350]°F) to MODE 5 (T_{cold} < [200]°F) during normal and post accident operations. The time required to cool from [350]°F to [200]°F is a function of the number of CCW and [DHR] trains operating.

BASES

APPLICABLE SAFETY ANALYSES (continued)

One CCW train is sufficient to remove decay heat during subsequent operations with T_{cold} < [200]°F.

The CCW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The CCW trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one train does not depend on the other. In the event of a DBA, one train of CCW is required to provide the minimum heat removal capability assumed in the safety analysis for systems to which it supplies cooling water. To ensure this is met, two CCW trains must be OPERABLE. At least one CCW train will operate assuming the worst case single active failure occurs coincident with loss of offsite power.

A CCW train is considered OPERABLE when:

a. It has an OPERABLE pump and associated surge tank and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of CCW from other components or systems not required for safety may render these components or systems inoperable, but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system that must be prepared to perform its post accident safety functions, primarily Reactor Coolant System heat removal, by cooling the DHR heat exchanger.

In MODES 5 and 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable CCW train results in an inoperable EDG or DHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

BASES

ACTIONS (continued)

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CCW train cannot be restored to OPERABLE status in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable, but does not affect the OPERABILITY of the CCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves which cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally

BASES

SURVEILLANCE REQUIREMENTS (continued)

operating system that cannot be fully actuated as part of routine testing during normal operation. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The [18] 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 the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The [18] 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 the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [9.2.2].

B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a transient or Design Basis Accident (DBA) or transient. During normal operation and normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related position is covered by this LCO.

An SWS consists of two separate, 100% capacity safety related cooling water trains. Each train consists of a 100% capacity pump, one component cooling water (CCW) heat exchanger, piping, valving, and instrumentation. The pumps and valves are remote manually aligned, except in the unlikely event of a loss of coolant accident (LOCA). The pumps are automatically started upon receipt of a safety feature actuation signal, and all essential valves are aligned to their post accident positions. The SWS also provides cooling directly to the Control Room Emergency Ventilation System water cooled condensing unit, the Emergency Core Cooling System (ECCS) pump room coolers, containment air cooler, and turbine driven cooling water systems. The system provides cooling and is also a source of water to the ECCS pump and the emergency feedwater pumps, and can provide a source of makeup water to the cooling tower.

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the FSAR, Section [9.2.1] (Ref. 1). The principal safety related function of the SWS is the removal of decay heat from the reactor via the [CCW System].

APPLICABLE SAFETY ANALYSES

The design basis of the SWS is for one SWS train, in conjunction with the CCW System and a 100% capacity containment cooling system, (containment spray, containment air coolers, or a combination) to remove core decay heat following a design basis LOCA, as discussed in the FSAR, Section [6.2] (Ref. 2). This provides for a gradual reduction in the temperature of this fluid, as it is supplied to the Reactor Coolant System (RCS) by the safety injection pumps.

The SWS is designed to perform its function with a single failure of any active component, assuming loss of offsite power.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SWS, in conjunction with the CCW System, also cools the unit from Decay Heat Removal (DHR) System, as discussed in the FSAR, Section [6.3], (Ref. 3) entry conditions to MODE 5 during normal and post accident operation. The time required for this evolution is a function of the number of CCW and DHR System trains that are operating. One SWS train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum SWS temperature of [85]°F occurring simultaneously with maximum heat loads on the system.

The SWS is also required when needed to support CCW in the removal of heat from the emergency diesel generators (EDGs) or reactor auxiliaries.

The SWS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two SWS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power.

An SWS train is considered OPERABLE when:

a. It has an OPERABLE pump and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SWS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports.

ACTIONS

A.1

If one SWS train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SWS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the

BASES

ACTIONS (continued)

OPERABLE SWS train could result in loss of SWS function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable SWS train results in an inoperable EDG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable SWS train results in an inoperable DHR train. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SWS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the isolation of the SWS components or systems may render those components inoperable but does not affect the OPERABILITY of the SWS.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

The SR verifies proper automatic operation of the SWS valves. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

The SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at an [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [9.2.1].
2. FSAR, Section [6.2].
3. FSAR, Section [6.3].

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for process and operating heat from safety related components during a transient or accident as well as during normal operation. This is done utilizing the Service Water System (SWS).

The UHS has been defined as that complex of water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures, as discussed in the FSAR, Section [9.2.5] (Ref. 1). If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink. The two principal functions of the UHS are the dissipation of residual heat after a reactor shutdown, and dissipation of residual heat after an accident.

A variety of complexes is used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required.

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. Basins of cooling towers generally include less than a 30 day supply of water, typically 7 days or less. A 30 day supply would be dependent on another source(s) and a makeup system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a 1 day supply, the systems for replenishing the basin and the backup source(s) become of sufficient importance that the makeup system itself may be required to meet the same design criteria as an Engineered Safety Feature (e.g., single failure considerations and multiple makeup water sources may be required).

Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1.

BASES

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removal from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on [decay heat removal]. Its maximum post accident heat load occurs approximately 20 minutes after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation and the containment cooling systems are required to remove the core decay heat.

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis. These assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case failure (e.g., single failure of a manmade structure). The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the UHS.

The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if [it contains a sufficient volume of water at or below the maximum temperature] that would allow the SWS to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed [90]°F, and the level should not fall below [562] ft [mean sea level] during normal unit operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and is required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

[A.1

If one or more cooling towers have one fan inoperable (i.e., up to one fan per cooling tower inoperable), action must be taken to restore the inoperable cooling tower fan(s) to OPERABLE status within 7 days.

The 7 day Completion Time is reasonable, based on the low probability of an accident occurring during the 7 days that one cooling tower fan is

BASES

ACTIONS (continued)

inoperable in one or more cooling towers, the number of available systems, and the time required to complete the Required Action.]

[B.1

- REVIEWER'S NOTE -

The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit.

With water temperature of the UHS > [90]°F, the design basis assumption associated with initial UHS temperature are bounded provided the temperature of the UHS averaged over the previous 24 hour period is ≤ [90]°F. With the water temperature of the UHS > [90]°F, long term cooling capability of the ECCS loads and DGs may be affected. Therefore, to ensure long term cooling capability is provided to the ECCS loads when water temperature of the UHS is > [90]°F, Required Action B.1 is provided to more frequently monitor the water temperature of the UHS and verify the temperature is ≤ [90]°F when averaged over the previous 24 hour period. The once per hour Completion Time takes into consideration UHS temperature variations and the increased monitoring frequency needed to ensure design basis assumptions and equipment limitations are not exceeded in this condition. If the water temperature of the UHS exceeds [90]°F when averaged over the previous 24 hour period or the water temperature of the UHS exceeds [ ]°F, Condition C must be entered immediately.]

[C.1 and C.2

If the Required Actions and Completion Time of Condition [A or B] are not met, or the UHS is inoperable [for reasons other than Condition A or B], the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.]

BASES

SURVEILLANCE REQUIREMENTS

[SR 3.7.9.1

This SR verifies that adequate long term (30 days) cooling can be maintained. The level specified also ensures NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ [ ] ft [mean sea level].]

[SR 3.7.9.2

This SR verifies that the SWS can cool the CCW System to at least its maximum design temperature within the maximum [accident or normal heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS average water temperature is ≤ [90]°F.]

[SR 3.7.9.3

Operating each cooling tower fan for ≥ [15] minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.]

REFERENCES

1. FSAR, Section [9.2.5].
2. Regulatory Guide 1.27.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity[, chemicals, or toxic gas].

The CREVS consists of two independent, redundant, fan filter assemblies. Each filter train consists of a roughing filter, a high efficiency particulate air (HEPA) filter, and a charcoal filter.

The CREVS is an emergency system. Upon receipt of the activating signal(s), the normal control room ventilation system is automatically shut down and the CREVS can be manually started. The roughing filters and water condensing units remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA and charcoal filters.

A single train will pressurize the control room with a 1.5 ft^2 LEAKAGE area to about 1/8 inch water gauge. The CREVS operation is discussed in the FSAR, Section [9.4] (Ref. 1).

The CREVS is designed to maintain the control room for 30 days of continuous occupancy after a Design Basis Accident (DBA), without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The CREVS components are arranged in redundant safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the control room operators as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident fission product release presented in the FSAR, Chapter [15] (Ref. 2).

The worst case single active failure of a CREVS component, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

[For this unit, there are no sources of toxic gases or chemicals that could be released to affect control room habitability.]

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CREVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant CREVS trains are required to be OPERABLE to ensure that at least one is available if a single failure disables the other train. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a large radioactive release.

The CREVS is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both trains. A CREVS train is considered OPERABLE when the associated:

a. Fan is OPERABLE,

b. HEPA filter and charcoal absorber are not excessively restricting flow, and are capable of performing their filtration functions, and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary, including the integrity of the walls, floors, ceilings, ductwork, and access doors, must be maintained within the assumptions of the design analysis.

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

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the control room will remain habitable during and following a DBA.

During movement of [recently] irradiated fuel assemblies, the CREVS must be OPERABLE to cope with a release due to a fuel handling accident [involving handling recently irradiated fuel. Due to radioactive decay, CREVS is only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [ ] days)].

BASES

ACTIONS

A.1

With one CREVS train inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the control room radiation protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1

- REVIEWER'S NOTE -

Adoption of Condition B is dependent on a commitment from the licensee to have written procedures available describing compensatory measures to be taken in the event of an intentional or unintentional entry into Condition B.

If the control room boundary is inoperable in MODE 1, 2, 3, or 4, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the control room boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train or control room boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

[D.1 and D.2

In MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREVS train must immediately be placed in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected. Required Action D.1 is modified by a Note indicating to place the system in the emergency mode if automatic transfer to emergency mode is inoperable.

An alternative to Required Action D.1 is to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.]

[E.1

In MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies, when two CREVS trains are inoperable, action must be taken immediately to suspend activities that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.]

F.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable control room boundary (i.e., Condition B), the CREVS may not be capable of performing the intended function and the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month adequately checks this system. Monthly heater operations dry out any moisture that has accumulated in the charcoal because of humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be

BASES

SURVEILLANCE REQUIREMENTS (continued)

operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.10.2

This SR verifies that the required CREVS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The [VFTP] includes testing HEPA filter performance, charcoal absorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.10.3

This SR verifies that [each CREVS train starts] [or the control room isolates] and operates on an actual or simulated actuation signal. The Frequency of [18] months is consistent with that specified in Reference 3.

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify that the CREVS is functioning properly. During the emergency mode of operation, the CREVS is designed to pressurize the control room ≥ [0.125] inches water gauge positive pressure, with respect to adjacent areas, to prevent unfiltered inleakage. The CREVS is designed to maintain this positive pressure with one train at a flow rate of ≤ [3300] cfm. This value includes [300] cfm of outside air. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs.

REFERENCES

1. FSAR, Section [9.4].
2. FSAR, Chapter [15].
3. Regulatory Guide 1.52, Rev. [2].

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

BASES

BACKGROUND

The CREATCS provides temperature control for the control room following isolation of the control room.

The CREATCS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment. Ductwork, valves or dampers, and instrumentation also form part of the system. Two redundant air cooled condensing units are provided as a backup to the water cooled condensing unit. Both the water cooled and air cooled condensing units must be OPERABLE for the CREATCS to be OPERABLE. During emergency operation, the CREATCS maintains the temperature between 70°F and 85°F. The CREATCS is a subsystem providing air temperature control for the control room.

The CREATCS is an emergency system. On detection of high containment building pressure or radiation, low Reactor Coolant System pressure, or high noble gas radioactivity in the station vent, the normal control room ventilation system is automatically shut down, and the Control Room Emergency Ventilation System can be manually started. A single train will provide the required temperature control. The CREATCS operation to maintain control room temperature is discussed in the FSAR, Section [9.4] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CREATCS is to maintain control room temperature for 30 days of continuous occupancy.

The CREATCS components are arranged in redundant, safety related trains. During emergency operation, the CREATCS maintains the temperature between [70]°F and [95]°F. A single active failure of a CREATCS component does not impair the ability of the system to perform as designed. The CREATCS is designed in accordance with Seismic Category I requirements. The CREATCS is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

The CREATCS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Two independent and redundant trains of the CREATCS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CREATCS is considered OPERABLE when the individual components that are necessary to maintain control room temperature are OPERABLE in both trains. These components include the cooling coils, water cooled condensing units, and associated temperature control instrumentation. In addition, the CREATCS must be OPERABLE to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, 4, [5, and 6,] and during movement of [recently] irradiated fuel assemblies [i.e., fuel that has occupied part of a critical reactor core within the previous [ ] days)], the CREATCS must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY requirements following isolation of the control room.

ACTIONS

A.1

With one CREATCS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREATCS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a failure in the OPERABLE CREATCS train could result in a loss of CREATCS function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining train can provide the required capabilities, and the alternate safety or nonsafety related cooling means that are available.

Concurrent failure of two CREATCS trains would result in the loss of function capability; therefore, LCO 3.0.3 must be entered immediately.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in 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

BASES

ACTIONS (continued)

unit conditions from full power conditions in an orderly manner without challenging unit systems.

[C.1 and C.2

[In MODE 5 or 6, or] during movement of [recently] irradiated fuel, if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREATCS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that could release radioactivity that might require the isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.]

[D.1

[In MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies, with two CREATCS trains inoperable, action must be taken to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.]

E.1

If both CREATCS trains are inoperable in MODE 1, 2, 3, or 4, the CREATCS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the [safety analyses]. This SR consists of a combination of testing and calculations. An [18] month Frequency is appropriate, as significant degradation of the CREATCS is slow and is not expected over this time period.

BASES

REFERENCES	1.	FSAR, Section [9.4].

B 3.7 PLANT SYSTEMS

B 3.7.12 Emergency Ventilation System (EVS)

BASES

BACKGROUND

The EVS filters air from the area of the active Emergency Core Cooling System (ECCS) components during the recirculation phase of a loss of coolant accident (LOCA).

The EVS consists of two independent, redundant trains. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the Auxiliary Building negative pressure area following receipt of a safety features actuation signal (SFAS).

The EVS is a standby system. During emergency operations, the EVS dampers are realigned, and fans are started to begin filtration. Upon receipt of the SFAS signal(s), normal air discharges from the negative pressure area are isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The EVS is discussed in the FSAR, Sections [6.2.3], [9.4.2], and 15.4.6] (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the EVS is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as an ECCS pump seal failure during the recirculation mode. In such a case, the system limits radioactive release to within 10 CFR 100 (Ref. 4) requirements. The analysis of the effects and consequences of a large break LOCA is presented in Reference 3. The EVS also actuates following a small break LOCA, in those cases where the unit goes into the recirculation mode of long term cooling, and to cleanup releases of smaller leaks, such as from valve stem packing.

Two types of system failures are considered in the accident analysis: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal of any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Following a LOCA, an ESFAS signal starts the EVS fans and opens the dampers located in the penetration room outlet ductwork. The ESFAS signal closes all containment isolation valves and purge system valves. The purge system fans, if running, are shut down automatically.

The EVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant trains of the EVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. Total system failure could result in atmospheric release from the negative pressure area boundary exceeding Reference 4 limits in the event of a Design Basis Accident (DBA).

The EVS is considered OPERABLE when the individual components necessary to maintain the negative pressure area boundary filtration are OPERABLE in both trains.

An EVS train is considered OPERABLE when its associated:

a. Fan is OPERABLE,

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions, and

c. [Heater, demister,] ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The LCO is modified by a Note allowing the Auxiliary Building negative pressure area 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 Auxiliary Building negative pressure area isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, the EVS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

BASES

APPLICABILITY (continued)

In MODES 5 and 6, the EVS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one EVS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the EVS safety function. However, the overall reliability is reduced because a single failure in the OPERABLE EVS train could result in loss of EVS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that of the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1

- REVIEWER'S NOTE -

Adoption of Condition B is dependent on a commitment from the licensee to have written procedures available describing compensatory measures to be taken in the event of an intentional or unintentional entry into Condition B.

If the Auxiliary building negative pressure area boundary is inoperable, the EVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE Auxiliary Building negative pressure area boundary within 24 hours. During the period that the Auxiliary Building negative pressure area boundary is inoperable, appropriate compensatory measures [consistent with the intent, as applicable, of GDC 19, 63, 64 and 10 CFR Part 100] should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the Auxiliary Building negative pressure area boundary.

BASES

ACTIONS (continued)

C.1 and C.2

If the EVS train or the Auxiliary Building negative pressure area boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations dry out any moisture that may have accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on known reliability of equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required EVS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.12.3

This SR verifies that each EVS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 5.

SR 3.7.12.4

This SR verifies the integrity of the negative pressure boundary area. The ability of the EVS to maintain a negative pressure, with respect to

BASES

SURVEILLANCE REQUIREMENTS (continued)

potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the EVS. During the [post accident] mode of operation, the EVS is designed to maintain a slight negative pressure in the negative pressure boundary area with respect to adjacent areas to prevent unfiltered LEAKAGE. The EVS is designed to maintain this negative pressure at a flow rate of [3000] cfm from the negative pressure boundary area. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs.

[SR 3.7.12.5

Operating the EVS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the EVS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with that specified in Reference 5.]

REFERENCES

1. FSAR, Section [6.2.3].
2. FSAR, Section [9.4.2].
3. FSAR, Section [15.4.6].
4. 10 CFR 100.11.
5. Regulatory Guide 1.52, Rev. [2].

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Storage Pool Ventilation System (FSPVS)

BASES

BACKGROUND

The FSPVS provides negative pressure in the fuel storage area, and filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident.

The FSPVS consists of portions of the normal Fuel Handling Area Ventilation System (FHAVS), the station Emergency Ventilation System (EVS), ductwork bypasses, and dampers. The portion of the normal FHAVS used by the FSPVS consists of ducting between the spent fuel pool and the normal FHAVS exhaust fans or dampers, and redundant radiation detectors installed close to the suction end of the FHAVS exhaust fan ducting. The portion of the EVS used by the FSPVS consists of two independent, redundant trains. Each train consists of a heater, prefilter, or high efficiency particulate air (HEPA) filter, activated charcoal adsorber section for removal of gaseous activity (principally iodines), and fan. Ductwork, valves or dampers, and instrumentation also form part of the system. Two isolation valves are installed in series in the ductwork between the FHAVS and the EVS to provide isolation of the EVS from the FHAVS on an Engineered Safety Feature actuation signal. These valves are opened prior to fuel handling operations [involving handling recently irradiated fuel]. The EVS is the subject of LCO 3.7.12, "Emergency Ventilation System (EVS)," and is fully described in the FSAR, Section [6.2.3], Reference 12. A ductwork bypass with redundant dampers connects the FHAVS to the EVS.

During normal operation, the exhaust from the fuel handling area is passed through the FHAVS exhaust filter and is discharged through the station vent stack. In the event of a fuel handling accident, the radiation detectors (one per EVS train), located at the suction of the FHAVS exhaust fan ducting, send signals to isolate the FHAVS supply and exhaust fans and ductwork, open the redundant dampers in the bypass ductwork, and start the EVS fans. The EVS fans pull the air from the fuel handling area, creating a negative pressure, and discharge the filtered air to the station vent.

The FHAVS is discussed in the FSAR, Sections [6.2.3], [9.4.2], and [15.4.7] (Refs. 1, 2, and 3, respectively), because it may be used for normal as well as post accident, atmospheric cleanup functions.

BASES

APPLICABLE SAFETY ANALYSES

The FSPVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident [involving handling recently irradiated fuel]. The analysis of the fuel handling accident, given in Reference 3, assumes that a certain number of fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident [involving handling recently irradiated fuel] assumes that only one train of the FSPVS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).

The FSPVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

[Two] independent and redundant trains of the FSPVS are required to be OPERABLE to ensure that at least one is available, assuming a single failure that disables the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the fuel handling area exceeding 10 CFR 100 (Ref. 5) limits in the event of a fuel handling accident [involving handling recently irradiated fuel].

The FSPVS is considered OPERABLE when the individual components necessary to control operator exposure in the fuel handling building are OPERABLE in both trains. An FSPVS train is considered OPERABLE when its associated:

1. Fan is OPERABLE,
2. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions, and
3. [Heater, demister,] ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The LCO is modified by a Note allowing the fuel building 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 fuel building isolation is indicated.

BASES

APPLICABILITY

[In [MODES 1, 2, 3, and 4,] the FSPVS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a loss of coolant accident (refer to LCO 3.7.12) for units that use this system as part of their EVSs.

During movement of [recently] irradiated fuel assemblies in the fuel handling area, the FSPVS is always required to be OPERABLE to mitigate the consequences of a fuel handling accident.

In MODES 5 and 6, the FSPVS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.]

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

With one FSPVS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time period, the remaining OPERABLE train is adequate to perform the FSPVS function. However, the overall reliability is reduced because a single failure in the OPERABLE FSPVS train could result in a loss of FSPVS functioning. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable FSPVS train, and ability of the remaining FSPVS train to provide the required protection.

B.1

- REVIEWER'S NOTE -

Adoption of Condition B is dependent on a commitment from the licensee to have written procedures available describing compensatory measures to be taken in the even of an intentional or unintentional entry into Condition B.

If the fuel building boundary is inoperable in MODE 1, 2, 3, or 4, the FSPVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE fuel building boundary within 24 hours.

BASES

ACTIONS (continued)

During the period that the fuel building boundary is inoperable, appropriate compensatory measures [consistent with the intent, as applicable, of GDC 19, 60, 61, 63, 64 and 10 CFR 100] should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the fuel building boundary.

[C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the associated Completion Time, or when both FSPVS trains are inoperable for reasons other than an inoperable fuel building boundary (i.e., Condition B), the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.]

D.1 and D.2

If the inoperable FSPVS train cannot be restored to OPERABLE status within the required Completion Time, during movement of [recently] irradiated fuel assemblies in the fuel building the OPERABLE FSPVS train must be started immediately or [recently] irradiated fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failures will be readily detected.

If the system is not placed in operation, this action requires suspension of [recently] irradiated fuel movement, which precludes a fuel handling accident [involving handling recently irradiated fuel]. This action does not preclude the movement of fuel assemblies to a safe position.

BASES

ACTIONS (continued)

E.1

When two trains of the FSPVS are inoperable during movement of [recently] irradiated fuel assemblies in the fuel building, the unit must be placed in a condition in which the LCO does not apply. This LCO involves immediately suspending movement of [recently] irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

[SR 3.7.13.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.]

[SR 3.7.13.2

This SR verifies that the required FSPVS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].]

[SR 3.7.13.3

This SR verifies that each FSPVS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with that specified in Reference 6.]

SR 3.7.13.4

This SR verifies the integrity of the fuel handling area. The ability of the fuel handling area to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify

BASES

SURVEILLANCE REQUIREMENTS (continued)

proper function of the FSPVS. During the [post accident] mode of operation, the FSPVS is designed to maintain a slight negative pressure in the fuel handling area to prevent unfiltered LEAKAGE. The FSPVS is designed to maintain this negative pressure at a flow rate of ≤ [3000] cfm to the fuel handling area. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice.

SR 3.7.13.5

Operating the FSPVS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FSPVS filter bypass damper is verified if it can be opened. A Frequency of [18] months is specified in Reference 6.

REFERENCES

1. FSAR, Section [6.2.3].
2. FSAR, Section [9.4.2].
3. FSAR, Section [15.4.7].
4. Regulatory Guide 1.25.
5. 10 CFR 100.11.
6. Regulatory Guide 1.52, Rev. [2].

B 3.7 PLANT SYSTEMS

B 3.7.14 Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the fuel storage pool meets the assumption of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the fuel storage pool design is given in the FSAR, Section [9.1.2], Reference 1. The Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section [9.1.3] (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section [15.4.7] (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is below 10 CFR 100 (Ref. 5) guidelines. According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With 23 ft, the assumptions of Reference 4 can be used directly. In practice, the LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel rack, however, there may be < 23 ft above the top of the fuel bundle and the surface, by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although the analysis shows that only the first [few] rows fail from a hypothetical maximum drop. The fuel storage pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool since the potential for a release of fission products exists.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for an accident cannot be met, immediate action must be taken to preclude the occurrence of an accident. With the fuel storage pool at less than the required level, the movement of fuel assemblies in the fuel storage pool is immediately suspended. This effectively precludes the occurrence of a fuel handling accident. In such a case, unit procedures control the movement of loads over the spent fuel. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.

During refueling operations, the level in the fuel storage pool is at equilibrium with that in the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.

REFERENCES

1. FSAR, Section [9.1.2].
2. FSAR, Section [9.1.3].
3. FSAR, Section [15.4.7].
4. Regulatory Guide 1.25.
5. 10 CFR 100.11.

B 3.7 PLANT SYSTEMS

B 3.7.15 [Spent Fuel Pool Boron Concentration]

BASES

BACKGROUND	As described in the following LCO 3.7.16, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel pool racks [in a "checkerboard" pattern] in accordance with criteria based on [initial enrichment and discharge burnup]. Although the water in the spent fuel pool is normally borated to ≥ [500] ppm, the criteria that limit the storage of a fuel assembly to specific rack locations are conservatively developed without taking credit for boron.
APPLICABLE SAFETY ANALYSES	A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.16 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). This accident is analyzed assuming the extreme case of completely loading the spent fuel pool racks with unirradiated assemblies of maximum enrichment. Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded spent fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The specified concentration [≤ [500] ppm] of dissolved boron in the fuel storage pool preserves the assumption used in the analyses of the potential accident scenarios described above. 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 pool, until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movement in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

BASES

ACTIONS	<u>A.1, A.2.1, and A.2.2</u> The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. 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 the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. The concentration of boron is restored simultaneously with suspending movement of the fuel assemblies. Alternatively, beginning a verification of the spent fuel pool locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. 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 operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	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 incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.
REFERENCES	None.

B 3.7 PLANT SYSTEMS

B 3.7.16 [Spent Fuel Pool Storage]

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store [735] fuel assemblies, which includes storage for [15] failed fuel containers. The spent fuel storage cells are installed in parallel rows with center to center spacing of [12 31/32] inches in one direction, and [13 3/16] inches in the other orthogonal direction. This spacing and "flux trap" construction, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans, is sufficient to maintain a k_{eff} of ≤ 0.95 for spent fuel of original enrichment of up to [3.3]%. However, as higher initial enrichment fuel assemblies are stored in the spent fuel pool, they must be stored in a checkerboard pattern taking into account fuel burnup to maintain a k_{eff} of 0.95 or less.

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.

The spent fuel pool storage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the fuel pool, according to Figure [3.7.16-1] in the accompanying LCO, ensure that the k_{eff} of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool, according to Figure [3.7.16-1]. Fuel assemblies not meeting the criteria of Figure [3.7.16-1] 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 the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure [3.7.16-1], immediate action must be taken

BASES

ACTIONS (continued)

to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure [3.7.16-1].

If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure [3.7.16-1] in the accompanying LCO. For fuel assemblies in the unacceptable range of [Figure 3.7.16-1], performance of the SR will ensure compliance with Specification 4.3.1.1.

REFERENCES

None.

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube out-LEAKAGE from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicative of current conditions. During transients, I-131 spikes have been observed, as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products, in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational Leakage") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant leakage. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.79 rem if the main steam safety valves (MSSVs) are open for the 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break, as discussed in the FSAR, Chapter [15] (Ref. 2) 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 an MSLB do not exceed established limits, (Ref. 1) for whole body and thyroid dose rates.

BASES

APPLICABLE SAFETY ANALYSES (continued)

With a loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Emergency Feedwater System supplies the necessary makeup to the steam generator. Venting continues until the reactor coolant temperature and pressure has decreased sufficiently for the Shutdown Cooling System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ [0.10] μCi/gm DOSE EQUIVALENT I-131 maintains the radiological consequences of a Design Basis Accident (DBA) to a small fraction of Reference 1 limits.

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.

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 at low pressure and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS	A.1 and A.2 DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.17.1 This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.
REFERENCES	1. 10 CFR 100.11. 2. FSAR, Chapter [15].

B 3.7 PLANT SYSTEMS

B 3.7.18 Steam Generator Level

BASES

BACKGROUND

A principal function of the steam generators is to provide superheated steam at a constant pressure (900 psia) over the power range. Steam generator water inventory is maintained large enough to provide adequate primary to secondary heat transfer. Mass inventory and indicated water level in the steam generator increases with load as the length of the four heat transfer regions within the steam generator vary. Inventory is controlled indirectly as a function of power and maintenance of a constant average primary system temperature by the feedwater controls in the Integrated Control System.

The maximum operating steam generator level is based primarily on preserving the initial condition assumptions for steam generator inventory used in the FSAR steam line break (SLB) analysis (Ref. 1). An inventory of 62,600 lb was used in this analysis. The 62,600 lb must not be exceeded due to the concerns of a possible return to criticality because of primary side cooling following an SLB and the maximum pressure in the reactor building.

For a clean once through steam generator, the mass inventory in a steam generator for operating at 100% power is approximately 39,000 lb to 40,000 lb.

As a steam generator becomes fouled and the operating level approaches the limit of 96%, the mass inventory in the downcomer region increases approximately 10,000 lb, and adds to the total mass inventory of the steam generator. In matching unit data of startup level versus power, the steam generator performance codes have shown that fouling of the lower tube support plates does not significantly change the heat transfer characteristics of the steam generator. Thus, the steam temperature, or superheat, is not degraded due to the fouling of the tube support plates, and mass inventory changes are mainly due to the added level in the downcomer.

Analytically, increasing the fouling of the steam generator tube surfaces degrades the heat transfer capability of the steam generator, increases the mass inventory, and decreases the steam superheat at 100% power (2544 MW). The results were presented as the amount of mass inventory in each steam generator versus operating range level and steam superheat.

BASES

BACKGROUND (continued)

The limiting curve, which was determined from several steam generator performance code runs at a power level of 100%, conservatively bounds steam generator mass inventory value, when operating at power levels < 100%.

The points displayed in Figure 3.7.18-1, in the accompanying LCO, are the intercept points of the 57,000 lb mass value, and the operating range level x and steam superheat values.

The steam generator performance analysis also indicated that startup and full range level instruments are inadequate indicators of steam generator mass inventory at high power levels due to the combination of static and dynamic pressure losses. If the water level should rise above the 96% upper limit, the steam superheat would tend to decrease due to reduced feedwater heating through the aspirator ports. Normally, a reduction in water level is manually initiated to maintain steam flow through the aspirator port by reducing the power level. Thus, the superheat versus level limitation also tends to ensure that, in normal operation, water level will remain clear of the aspirator ports.

Feedwater nozzle flooding would impair feedwater heating, and could result in excessive tube to shell temperature differentials, excessive tubesheet temperature differentials, and large variations in pressurizer level.

APPLICABLE SAFETY ANALYSES

The most limiting Design Basis Accident that would be affected by steam generator operating level is a steam line failure. This accident is evaluated in Reference 1. The parameter of interest is the mass of water, or inventory, contained in the steam generator due to its role in lowering Reactor Coolant System (RCS) temperature (return to criticality concern), and in raising containment pressure during an SLB accident. A higher inventory causes the effects of the accident to be more severe. Figure 3.7.18-1, in the accompanying LCO, is based upon maintaining inventory < 57,000 lb, which is 10% less than the inventory used in the FSAR accident analysis, and therefore is conservative.

The steam generator level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO is required to preserve the initial condition assumptions of the accident analyses. Failure to meet the maximum steam generator level LCO requirements can result in additional mass and energy released to containment, and excessive cooling (and related core reactivity effects)

BASES

LCO (continued)

following an SLB. In addition, feedwater nozzle flooding would impair feedwater heating, and could result in excessive tube to shell temperature differentials and excessive tubesheet temperature gradients.

APPLICABILITY

In MODES 1 and 2, a maximum steam generator water level is required to preserve the initial condition assumption for steam generator inventory used in the steam line failure accident analysis (Ref. 1).

In MODE 3, limits on RCS boron concentrations will prevent a return to criticality in the event of an SLB. In MODES 4, 5, and 6, the water in the steam generator has a low specific enthalpy; therefore, there is no need to limit the steam generator inventory when the unit is in this condition.

ACTIONS

A.1

With the steam generator level in excess of the maximum limit, action must be taken to restore the level to within the bounds assumed in the analysis. To achieve this status, the water level is restored to within the limit. The 15 minute Completion Time is considered to be a reasonable time to perform this evolution.

B.1

If the water level in one or more steam generators cannot be restored to less than or equal to the maximum level in Figure 3.7.18-1, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours. The allowed Completion Time is 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 3.7.18.1

This SR verifies the steam generator level to be within acceptable limits. The 12 hour Frequency is adequate because the operator will be aware of unit evolutions that can affect the steam generator level between checks. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to steam generator level status.

BASES

REFERENCES	1. FSAR, Section [15.4.4].