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# **Standard Technical Specifications Babcock and Wilcox Plants**

Bases (Sections 3.4–3.9)

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**Issued by the  
U.S. Nuclear Regulatory Commission**

**Office of Nuclear Reactor Regulation**

**April 1995**





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## PREFACE

This NUREG contains the improved Standard Technical Specifications (STS) for Babcock and Wilcox (B&W) plants. Revision 1 incorporates the cumulative changes to Revision 0, which was published in September 1992. The changes reflected in Revision 1 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This NUREG is the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, specifically the B&W Owners Group (BWOG), NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132). Licensees are encouraged to upgrade their technical specifications consistent with those criteria and conforming, to the extent practical and consistent with the licensing basis for the facility, to Revision 1 to the improved STS. The Commission continues to place the highest priority on requests for complete conversions to the improved STS. Licensees adopting portions of the improved STS to existing technical specifications should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.



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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

#### BASES

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##### 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.

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##### 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  $\geq [1.3]$ . This is the acceptance

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 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 the NRC Policy Statement.

---

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

(continued)

## BASES

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LCO  
(continued)      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.

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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.

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ACTIONS      A.1

Loop pressure and hot leg coolant temperature are controllable and measurable parameters. With one or both of

(continued)

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BASES

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ACTIONS

A.1 (continued)

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 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.

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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

(continued)



BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.1.1 (continued)

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.

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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.1.4 (continued)

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  $\Delta T$  across the core will be too small to provide valid results.

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REFERENCES

1. FSAR, Chapter [15].
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.2 RCS Minimum Temperature for Criticality

#### BASES

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##### 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;
- 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 the NRC Policy Statement.

---

##### 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).

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(continued)

BASES (continued)

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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  $k_{eff} \geq 1.0$ .

---

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 3 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.

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SURVEILLANCE  
REQUIREMENTS      SR 3.4.2.1

$T_{avg}$  is required to be verified above 525°F every 30 minutes. The 30 minute time period is frequent enough to prevent inadvertent violation of the LCO. The 30 minute portion of the Frequency has been modified by a Note indicating this SR is only required when  $T_{avg} < 530^\circ\text{F}$ . While Surveillance is required whenever the reactor is critical and temperature is below 530°F, in practice the Surveillance is most appropriate during the period when the reactor is brought critical.

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REFERENCES        1. FSAR, Chapter [15].

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.3 RCS Pressure and Temperature (P/T) Limits

#### BASES

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##### 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

(continued)

## BASES

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### BACKGROUND (continued)

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 (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.

(continued)

## BASES

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### BACKGROUND (continued)

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

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### 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 the NRC Policy Statement.

---

### 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

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## BASES

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### LCO (continued)

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;
- 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.

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(continued)



BASES (continued)

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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 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.

(continued)

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BASES

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ACTIONS  
(continued)

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be 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 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.

(continued)

## BASES

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### ACTIONS

#### C.1 and C.2 (continued)

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.

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### 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.

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.3.1 (continued)

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

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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.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.4 RCS Loops—MODES 1 and 2

#### BASES

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##### 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.

(continued)

## BASES

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### BACKGROUND (continued)

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.

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### 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

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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.

RCS Loops—MODES 1 and 2 satisfy Criterion 2 of the NRC Policy Statement.

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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.

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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).
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(continued)

BASES (continued)

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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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.4.1

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.

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REFERENCES

1. FSAR, Chapter [ ].
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.5 RCS Loops—MODE 3

#### BASES

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##### 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.

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##### 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.

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

RCS Loops—MODE 3 satisfy Criterion 3 of the NRC Policy Statement.

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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 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 be de-energized for  $\leq 8$  hours per 24 hour period for the transition to or from the Decay Heat Removal (DHR) System, and otherwise may be de-energized for  $\leq 1$  hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, boron reduction is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least  $[10]^{\circ}\text{F}$  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.

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(continued)

BASES (continued)

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

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ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the required 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 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 no RCS loop is OPERABLE or in operation, except as provided in the Note in the LCO section, all operations involving a reduction of RCS boron concentration must be immediately suspended. This is necessary because boron

(continued)

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## BASES

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### ACTIONS C.1 and C.2 (continued)

dilution requires forced circulation for proper homogenization. Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to operation and to OPERABLE status. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

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### 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 the required number of RCPs are 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 the required pump that is not in operation. 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.

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### REFERENCES None.

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.6 RCS Loops—MODE 4

#### BASES

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##### 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.

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##### APPLICABLE SAFETY ANALYSES

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

RCS Loops—MODE 4 have been identified in the NRC Policy Statement as an important contributor to risk reduction.

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##### 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 be de-energized for  $\leq 8$  hours per 24 hour period for the transition to or from the DHR System and otherwise may be de-energized for  $\leq 1$  hour per 8 hour period. This means that natural circulation has been

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BASES

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LCO  
(continued)

established using the SGs. The Note prohibits boron dilution 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.

The Note also permits the DHR pumps to be stopped for  $\leq 1$  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.

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(continued)

BASES (continued)

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

B.1 and B.2

If only one DHR loop is operable, an inoperable RCS or DHR loop must be restored to OPERABLE status to satisfy single failure considerations. The action must be started immediately and the immediate Completion Time reflects the urgency of restoring redundancy for heat removal. One loop is still available for cooldown for the reduced heat loads of this operating MODE.

If restoration cannot be 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 ( $\leq 200^{\circ}\text{F}$ ) rather than MODE 4 ( $200^{\circ}\text{F}$  to  $300^{\circ}\text{F}$ ). The Completion Time of 24 hours is reasonable, based on

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## BASES

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### ACTIONS B.1 and B.2 (continued)

operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems.

### C.1 and C.2

If no RCS or DHR loops are OPERABLE or 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.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.6.1

This Surveillance requires verification every 12 hours of the required number of DHR or RCS loops 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 the 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 the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls and has been shown to be acceptable by operating experience.

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(continued)



BASES (continued)

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REFERENCES       None.

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.7 RCS Loops—MODE 5, Loops Filled

#### BASES

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#### 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

(continued)

BASES

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BACKGROUND            removal path, the option to increase RCS pressure and  
(continued)            temperature for heat removal in MODE 4 is provided.

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APPLICABLE            No safety analyses are performed with initial conditions in  
SAFETY ANALYSES      MODE 5.

RCS Loops—MODE 5 (Loops Filled) have been identified in the  
NRC Policy Statement as important contributors to risk  
reduction.

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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  $\geq$  [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  
 $\geq$  [50]%. Should the operating DHR loop fail, the SGs could  
be used to remove the decay heat.

Note 1 permits the DHR pumps to be stopped 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 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.

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## BASES

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### LCO (continued)

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.

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 removal of DHR loops from 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.

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### 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";

(continued)

BASES

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APPLICABILITY (continued)	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).
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ACTIONS

A.1 and A.2

If one DHR loop is inoperable and any SG has secondary side water level < [50]%, redundancy for heat removal is lost. Action must be initiated to restore a second 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 A.1 or Required Action A.2 will restore redundant decay heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no DHR loop is in operation, except as provided in Note 1, or no required DHR loop is OPERABLE, all operations involving the reduction of RCS boron concentration must be suspended and action to restore a 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 Time reflects the importance of maintaining operation for decay heat removal.

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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.

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are  $\geq$  [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 the second 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 maintain decay heat removal and reactor coolant circulation. If the secondary side water level is  $\geq$  [50]% in both SGs, this Surveillance is not needed. Verification is performed by verifying proper breaker alignment and power available to the required pumps. 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.

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REFERENCES

None.

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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.8 RCS Loops—MODE 5, Loops Not Filled

#### BASES

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##### 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.

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##### 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) have been identified in the NRC Policy Statement as important contributors to risk reduction.

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(continued)

BASES (continued)

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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 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 be de-energized for  $\leq 15$  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  $\leq [160]^{\circ}\text{F}$ ]. The Note prohibits boron dilution 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.

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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).

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(continued)



BASES (continued)

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ACTIONS

A.1

If only one DHR loop is OPERABLE, 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 reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If both required loops are inoperable 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 boron reduction 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. 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification every 12 hours that at least one 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 the required number of pumps are OPERABLE ensures that redundancy for heat removal is provided. The requirement also ensures that additional loops 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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.8.2 (continued)

the required pumps. 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.

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REFERENCES

1. Generic Letter 88-17, October 17, 1988.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.9 Pressurizer

#### BASES

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##### 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 surge) 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

(continued)

## BASES

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### BACKGROUND (continued)

relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer surge 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.

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.

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### 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 noncondensable 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

(continued)

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

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 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 the NRC Policy Statement. 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.

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### LCO

The LCO requirement for the pressurizer to be OPERABLE with a water level  $\leq$  [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.

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## BASES

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### LCO (continued)

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.

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### APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus Applicability has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3 and, for pressurizer water level, for MODE 4 with RCS temperature  $\geq [275]^{\circ}\text{F}$ . 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  $[275]^{\circ}\text{F}$  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  $[275]^{\circ}\text{F}$ . 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

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## BASES

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APPLICABILITY (continued)	System is in service, and therefore the LCO is not applicable.
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## 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 temperature  $\leq [275]^{\circ}\text{F}$  places the plant into a MODE where the LCO is not applicable. The 12 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.

(continued)

BASES

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ACTIONS  
(continued)

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.

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

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

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.

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REFERENCES

1. NUREG-0737, November 1980.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.10 Pressurizer Safety Valves

#### BASES

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#### 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." For these conditions, the American Society of Mechanical Engineers (ASME) requirements are satisfied with one safety valve.

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  $\pm$  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  $\pm$  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.

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(continued)

BASES (continued)

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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, 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 the NRC Policy Statement.

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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  $\pm 1\%$  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.

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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

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BASES

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APPLICABILITY  
(continued)

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  $\leq [283]^{\circ}\text{F}$  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 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.

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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  $\leq [283]^{\circ}\text{F}$  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 12 hours allowed is

(continued)

BASES

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ACTIONS

B.1 and B.2 (continued)

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.

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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  $\pm$  [3]% for OPERABILITY; however, the valves are reset to  $\pm$  1% during the Surveillance to allow for drift.

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REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III, Section XI.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.11 Pressurizer Power Operated Relief Valve (PORV)

#### BASES

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##### 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.

(continued)

## BASES

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### BACKGROUND (continued)

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.

The primary purpose of this LCO is to ensure that the PORV, its setpoint, 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.

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### 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.

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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.

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 do not satisfy any specific Criterion of the NRC Policy Statement. This Specification was evaluated using insights gained from reviewing representative probabilistic risk assessments. The PORV and its block valve are deemed important to risk.

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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.

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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

(continued)



BASES

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APPLICABILITY  
(continued)

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.

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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.

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(continued)

BASES

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ACTIONS  
(continued)

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.

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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.

SR 3.4.11.3

This Surveillance is not required for plants with permanent IE 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.

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(continued)

BASES (continued)

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- 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.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

#### BASES

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#### 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.

(continued)

## BASES

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### BACKGROUND (continued)

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).

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 surge 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.

### PORV Requirements

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.

(continued)

## BASES

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### BACKGROUND (continued)

#### RCS Vent Requirements

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 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.

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### 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

(continued)

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

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.

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.

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

This LCO will deactivate the HPI actuation when the RCS temperature is  $\leq [283]^{\circ}\text{F}$ . 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.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORV is set to open at  $\leq [555]$  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  $\leq [220]$  inches to provide the 10 minute action time for correcting transients.

(continued)



BASES

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APPLICABLE  
SAFETY ANALYSES

Pressurizer Level Performance (continued)

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.

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 the NRC Policy Statement.

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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 only [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 pressurizer is OPERABLE with a coolant level  $\leq$  [220] inches.

The PORV is OPERABLE when its block valve is open, its lift setpoint is set at  $\leq$  [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.

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## BASES

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

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APPLICABILITY      This LCO is applicable in MODE 4 when any RCS cold leg temperature is  $\leq$  [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 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.

The Applicability is modified by a Note stating 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.

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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.

Required Action A.1 is modified by a Note that permits two pumps capable of RCS injection for  $\leq$  15 minutes to allow for pump swaps.

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BASES

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ACTIONS  
(continued)

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  $> 175^{\circ}\text{F}$ , 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.

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.

(continued)

## BASES

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### ACTIONS (continued)

#### 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.

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, the system must be restored to OPERABLE status within 1 hour. When this is not possible, 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  $\geq [0.75]$  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.

(continued)

## BASES

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### ACTIONS

#### I.1 and I.2 (continued)

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.

Again, 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 transient is not likely in those times.

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### 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.

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  $\leq$  [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

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.12.4 (continued)

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

When stipulated by LCO 3.4.12.b, the RCS vent of at least [0.75] square inches must be verified open for relief protection. For a vent valve not locked open, the Frequency is every 12 hours. For a valve locked open, 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.

A Note modifies the SR by requiring the Surveillance 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  $\leq$  [283]°F and every 31 days thereafter to ensure the setpoint is proper for

(continued)

## BASES

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### SURVEILLANCE REQUIREMENTS

#### SR 3.4.12.7 (continued)

using the PORV for LTOP. 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.

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### REFERENCES

1. 10 CFR 50, Appendix G.
  2. Generic Letter 88-11.
  3. FSAR, Section 15.
  4. 10 CFR 50.46.
  5. 10 CFR 50, Appendix K.
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.13 RCS Operational LEAKAGE

#### BASES

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##### 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,

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## BASES

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### BACKGROUND (continued)

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.

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### 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 the NRC Policy Statement.

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### 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

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BASES

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LCO  
(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.

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 identified 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.

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(continued)

BASES (continued)

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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.

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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.

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(continued)

BASES (continued)

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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 and near operating pressure. Therefore, this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady state operation near operating pressures have been established.

Steady state operation is required to perform a proper water inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. 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. The Note states that the SR is required to be performed in steady state operation.

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

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.13.2 (continued)

Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
  2. Regulatory Guide 1.45, May 1973.
  3. FSAR, Chapter [15].
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

#### BASES

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##### 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.

(continued)

## BASES

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### BACKGROUND (continued)

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;
- 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.

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### 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 the NRC Policy Statement.

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(continued)

BASES (continued)

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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.

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.

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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.

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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.

(continued)

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BASES

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ACTIONS  
(continued)

The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

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 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 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.

(continued)

## BASES

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### ACTIONS

#### B.1 and B.2 (continued)

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.

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### 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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.14.1 (continued)

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 resealed. Within 24 hours is a reasonable and practical time limit for performing this test after opening or resealing 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 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 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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.14.1 (continued)

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.

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REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 55a(c).

(continued)

BASES

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REFERENCES  
(continued)

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).
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.15 RCS Leakage Detection Instrumentation

#### BASES

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#### 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  $10^{-9}$   $\mu\text{Ci/cc}$  radioactivity for particulate monitoring and of  $10^{-6}$   $\mu\text{Ci/cc}$  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  $1^{\circ}\text{F}$  increase in dew

(continued)

## BASES

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### BACKGROUND (continued)

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 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.

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### 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 the NRC Policy Statement.

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(continued)

BASES (continued)

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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.

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.

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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  $\leq 200^{\circ}\text{F}$  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.

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ACTIONS                A.1 and A.2

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

However, the containment atmosphere 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.

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

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BASES

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ACTIONS

A.1 and A.2 (continued)

acceptable considering the frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

Required Action A.1 and Required Action A.2 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 monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.

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. The 30 day Completion Time recognizes at least one other form of leak detection is available.

Required Actions B.1.1, B.1.2, and B.2 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 containment atmosphere radioactivity monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.

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

(continued)

BASES

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ACTIONS

C.1 and C.2 (continued)

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.

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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. 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.

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.15.3 and SR 3.4.15.4 (continued)

Again, operating experience has proven this Frequency is acceptable.

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REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
  2. Regulatory Guide 1.45.
  3. FSAR, Section [ ].
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## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.16 RCS Specific Activity

#### BASES

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##### 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.

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##### 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

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 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 the NRC Policy Statement.

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LCO

The specific iodine activity is limited to 1.0  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131, and the gross specific activity in the primary coolant is limited to the number of  $\mu\text{Ci/gm}$  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.

(continued)

BASES

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LCO  
(continued)      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.

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APPLICABILITY      In MODES 1 and 2, and in MODE 3 with RCS average temperature  $\geq 500^{\circ}\text{F}$ , operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature  $< 500^{\circ}\text{F}$ , 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.

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ACTIONS      A Note to the ACTIONS 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.

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.

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BASES

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ACTIONS

A.1 and A.2 (continued)

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.

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 and C.2

With the gross specific activity in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.

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.

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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.

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.16.1 (continued)

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  $\geq 15\%$  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 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.

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(continued)



BASES (continued)

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- REFERENCES
1. 10 CFR 100.11.
  2. FSAR, Section [15.6.3].
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.1 Core Flood Tanks (CFTs)

#### BASES

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#### 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. Additionally, the valves are interlocked with RCS pressure to ensure that they will open automatically as RCS pressure is increased above

(continued)

## BASES

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### BACKGROUND (continued)

CFT pressure and to prevent inadvertent closure prior to an accident. The valves also receive an Engineered Safety Feature Actuation System (ESFAS) signal to open. These features ensure that the valves meet the requirement of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 for "operating bypasses" and that the CFTs will be available for injection without reliance on operator action.

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 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.

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### 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.

(continued)

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

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.

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  $\leq 0.17$  times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium water reaction of  $\leq 0.01$  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

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 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

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 the NRC Policy Statement.

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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.

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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  $\geq$  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.

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BASES

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APPLICABILITY  
(continued)

In MODE 3 with RCS pressure  $\leq$  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.

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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.

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

(continued)

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BASES

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ACTIONS

C.1 and C.2 (continued)

status, the plant must be brought to at least MODE 3 within 6 hours and RCS pressure reduced to  $\leq 750$  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.

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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.

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)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 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  $\geq$  [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. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves. Should closure of a valve occur, in spite of the interlock, the ESFAS signal provided to the valves would open a closed valve in the event of a LOCA].

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BASES (continued)

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- REFERENCES
1. FSAR, Section [6.3].
  2. 10 CFR 50.46.
  3. Draft NUREG-1366, February 1990.
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.2 ECCS—Operating

#### BASES

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##### 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

(continued)

BASES

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BACKGROUND  
(continued)

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 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.

(continued)

BASES

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BACKGROUND  
(continued)

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).

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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  $\leq 2200^{\circ}\text{F}$ ;
- b. Maximum cladding oxidation is  $\leq 0.17$  times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium water reaction is  $\leq 0.01$  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

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 uncover 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.

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 the NRC Policy Statement.

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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

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## BASES

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### LCO (continued)

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.

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### 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."

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BASES (continued)

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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. The intent of this Condition is to maintain a combination of equipment such that 100% of the safety injection flow equivalent to 100% of a single train remains available. 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 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

(continued)

BASES

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ACTIONS

B.1 and B.2 (continued)

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.

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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.

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

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.2.2 (continued)

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 noncondensable 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 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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.2.5 and SR 3.5.2.6 (continued)

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

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(continued)

BASES (continued)

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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 IWB-3400.
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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.3 ECCS—Shutdown

#### BASES

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**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.

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**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.

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**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,

(continued)

BASES

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LCO  
(continued)      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 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 a Note that 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.

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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."

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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

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BASES

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ACTIONS

A.1 (continued)

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 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.

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(continued)



BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply. This SR is modified by a Note that allows a DHR train to be considered OPERABLE during alignment and operation for DHR, 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, if necessary.

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REFERENCES

The applicable references from Bases 3.5.2 apply.

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## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.4 Borated Water Storage Tank (BWST)

#### BASES

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##### 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.

(continued)

## BASES

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BACKGROUND (continued)	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.
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APPLICABLE SAFETY ANALYSES	<p>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.</p>
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The limits on volume of [ $\geq 415,200$  gallons and  $\leq 449,000$  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

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 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.

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 the NRC Policy Statement.

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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 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.

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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."

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BASES (continued)

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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.

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.

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(continued)

BASES (continued)

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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.

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REFERENCES

1. FSAR, Section [6.1].
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.1 Containment

#### BASES

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##### 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 Accident (DBA). 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 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 (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

(continued)

BASES

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BACKGROUND  
(continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";
- b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";
- 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.

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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 loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. 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 (Ref. 1), as  $L_a$ : the maximum allowable leakage rate at the calculated maximum peak containment pressure ( $P_a$ ) resulting from the limiting DBA. The allowable leakage rate represented by  $L_a$  forms the basis for the acceptance criteria imposed on 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 the NRC Policy Statement.

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BASES (continued)

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LCO                      Containment OPERABILITY is maintained by limiting leakage to  $\leq 1.0 L_a$ , except prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test. At this time, the combined Type B and C leakage must be  $< 0.6 L_a$ , and the overall Type A leakage must be  $< 0.75 L_a$ . 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 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 acceptance criteria of Appendix J.

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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."

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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.

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BASES

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ACTIONS  
(continued)

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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions. Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test is required to be  $< 0.6 L_a$  for combined Type B and C leakage, and  $< 0.75 L_a$  for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of  $\leq 1.0 L_a$ . At  $\leq 1.0 L_a$  the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.1.2 (continued)

Frequency are consistent with the recommendations of  
Regulatory Guide 1.35 (Ref. 4).

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REFERENCES

1. 10 CFR 50, Appendix J.
  2. FSAR, Sections [14.1 and 14.2].
  3. FSAR, Section [5.6].
  4. Regulatory Guide 1.35, Revision [1].
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.2 Containment Air Locks

#### BASES

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##### 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.

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BASES (continued)

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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 (Ref. 1), as  $L_a$ : the maximum allowable containment leakage rate at the calculated maximum peak containment pressure ( $P_a$ ) following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air 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 the NRC Policy Statement.

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LCO

Each containment air lock forms part of the containment pressure boundary. As a part of containment, 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 and exit from containment.

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(continued)

BASES (continued)

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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."

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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 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."

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BASES

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ACTIONS  
(continued)

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 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

(continued)

BASES

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ACTIONS

A.1, A.2, and A.3 (continued)

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.

B.1, B.2, and B.3

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 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

(continued)

BASES

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ACTIONS

B.1, B.2, and B.3 (continued)

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 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,

(continued)

BASES

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ACTIONS

D.1 and D.2 (continued)

based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions. 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 Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

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 of SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the overall 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

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.2.2 (continued)

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 only challenged when the containment air lock door is opened, this test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently than every 184 days. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

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REFERENCES

1. 10 CFR 50, Appendix J.
  2. FSAR, Sections [14.1 and 14.2].
  3. FSAR, Section [5.6].
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.3 Containment Isolation Valves

#### BASES

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##### 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.

(continued)

BASES

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BACKGROUND  
(continued)

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 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<sup>3</sup>/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,

(continued)

BASES

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BACKGROUND            these valves may be opened as needed in MODES 1, 2, 3,  
(continued)            and 4.

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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.

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.

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 the NRC Policy Statement.

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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.

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.

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BASES (continued)

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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."

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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.

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

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BASES

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ACTIONS                    into the applicable Conditions and Required Actions of  
(continued)               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 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

(continued)

BASES

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ACTIONS

A.1 and A.2 (continued)

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 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 a Note that 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. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

B.1

With two containment isolation valves in one or more penetration flow paths inoperable (except for purge valve leakage not within limit), the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are

(continued)

BASES

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ACTIONS

B.1 (continued)

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 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 [4] 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. This Note is necessary since this Condition is

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BASES

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ACTIONS

C.1 and C.2 (continued)

written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by a Note that 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. 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

(continued)

BASES

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ACTIONS

D.1, D.2, and D.3 (continued)

not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the 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. 7). 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.

E.1 and E.2

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

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.3.1

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 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

(continued)

BASES

SURVEILLANCE  
REQUIREMENTS

SR 3.6.3.1 (continued)

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. 6), 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 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

(continued)



BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.3.3 (continued)

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 controls are not required to meet the SR during the time the valves are open.

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 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.

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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.3.4 (continued)

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 power operated and automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. [The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program 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, 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. 7).

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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.3.7 (continued)

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.

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  $\leq$  [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 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.

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REFERENCES

1. 10 CFR 20.
2. FSAR, Section [5.6].
3. FSAR, Sections [14.1 and 14.2].

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BASES

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REFERENCES  
(continued)

4. FSAR, Section [5.3].
  5. FSAR, Section [5.3].
  6. Generic Issue B-24.
  7. Generic Issue B-20.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.4 Containment Pressure

#### BASES

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**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.

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**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

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 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 the NRC Policy Statement.

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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.

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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.

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(continued)

BASES (continued)

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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 operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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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.

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REFERENCES

1. FSAR, Section [14.2].
  2. 10 CFR 50, Appendix K.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.5 Containment Air Temperature

#### BASES

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##### 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.

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##### 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

(continued)



BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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.

Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.

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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.

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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

(continued)

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BASES

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ACTIONS                    B.1 and B.2 (continued)

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.

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SURVEILLANCE  
REQUIREMENTS            SR 3.6.5.1

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

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REFERENCES                None.

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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.6 Containment Spray and Cooling Systems

#### BASES

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##### 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.

##### Containment Spray System

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

(continued)

## BASES

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### BACKGROUND

#### Containment Spray System (continued)

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 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

(continued)

BASES

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BACKGROUND

Containment Cooling System (continued)

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.

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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

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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.

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 the NRC Policy Statement.

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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.

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BASES

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LCO  
(continued)

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.

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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.

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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

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BASES

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ACTIONS

A.1 (continued)

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.

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.

(continued)



BASES

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ACTIONS  
(continued)

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.

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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

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.6.1 (continued)

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  $\geq 15$  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  $\geq [1780]$  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

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.6.4 (continued)

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 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.

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

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.

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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.
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.7 Spray Additive System

#### BASES

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##### 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.

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BASES (continued)

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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 the NRC Policy Statement.

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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

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## BASES

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LCO  
(continued)      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.

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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.

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## 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

(continued)

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BASES

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ACTIONS

B.1 and B.2 (continued)

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.

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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. 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.

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

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.

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].

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REFERENCES

1. FSAR, Section [6.2].
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.8 Hydrogen Recombiners

#### BASES

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##### 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.

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##### 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:

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

- 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;
- 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 the NRC Policy Statement.

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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.

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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

(continued)

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BASES

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APPLICABILITY (continued)      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.

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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.

(continued)

BASES

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ACTIONS                      B.1 and B.2 (continued)

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. [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.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.8.1

Performance of a system functional test for each hydrogen recombinder 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  $\geq 700^{\circ}\text{F}$  in  $\leq 90$  minutes. After reaching  $700^{\circ}\text{F}$ , the power is increased to maximum for approximately 2 minutes and power verified to be  $\geq 60$  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.

SR 3.6.8.2

This SR ensures that there are no physical problems that could affect recombinder 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  $\geq 10,000$  ohms. The [18] month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

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REFERENCES

1. Regulatory Guide 1.7, Revision [1].
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## B 3.7 PLANT SYSTEMS

### B 3.7.1 Main Steam Safety Valves (MSSVs)

#### BASES

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##### 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.

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##### APPLICABLE SAFETY ANALYSES

The design basis of the MSSVs comes from Reference 2 and its purpose is to limit secondary system pressure to  $\leq 110\%$  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;

(continued)

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

- b. Steam line break;
- c. Steam generator tube rupture; and
- d. Excessive heat removal due to feedwater system malfunction.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

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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 reseal 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.

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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

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## BASES

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APPLICABILITY (continued)      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.

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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:

$$RP = \frac{Y}{Z} \times 100\%$$

and

$$SP = \frac{Y}{Z} \times W$$

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];

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## BASES

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### ACTIONS

#### A.1 and A.2 (continued)

- 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 8 hours is allowed in Required Action A.2 to reduce the setpoints in recognition of the difficulty of resetting of all channels of this trip function within a period of 4 hours. The Completion Time of 12 hours for Required Action A.2 is based on 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

(continued)

BASES

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ACTIONS

B.1 and B.2 (continued)

experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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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.

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  $\pm [3]\%$  setpoint tolerance for OPERABILITY; however, the valves are reset to  $\pm 1\%$  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.

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(continued)

BASES (continued)

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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.
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## B 3.7 PLANT SYSTEMS

### B 3.7.2 Main Steam Isolation Valves (MSIVs)

#### BASES

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##### 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).

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##### 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.

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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.

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.

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

- 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 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 the NRC Policy Statement.

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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).

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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.

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## BASES

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### APPLICABILITY (continued)

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.

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### 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 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

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## BASES

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### ACTIONS

#### C.1 and C.2 (continued)

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.

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### SURVEILLANCE REQUIREMENTS

#### SR 3.7.2.1

This SR verifies that MSIV closure time of each MSIV is  $\leq$  [6] seconds on an actual or simulated actuation signal. The MSIV closure 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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.2.1 (continued)

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 or [18] months]. The [18] month Frequency to demonstrate the valve closure time 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, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3, with the unit at operating temperature and pressure, as discussed in the Reference 5 exercising requirements. 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.

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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.
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## 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

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##### 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).

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##### 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

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 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 the NRC Policy Statement.

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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.

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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.

(continued)

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BASES

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APPLICABILITY (continued)      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.

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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 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.

(continued)

BASES

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ACTIONS  
(continued)

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 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

(continued)

BASES

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ACTIONS                    C.1 and C.2 (continued)

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.

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.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each [MFSV], [MFCV], and [associated SFCV] is  $\leq 7$  seconds on an actual or simulated actuation signal.

The [MFSV], [MFCV], and [associated SFCV] closure 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 or [18] months]. The Frequency of [18] months for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency.

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REFERENCES

1. FSAR, Section [10.4.7].
  2. ASME, Boiler and Pressure Vessel Code, Section XI.
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## B 3.7 PLANT SYSTEMS

### B 3.7.4 Atmospheric Vent Valves (AVVs)

#### BASES

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##### 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.

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##### 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

(continued)

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

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) 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 the NRC Policy Statement.

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### 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.

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(continued)

BASES (continued)

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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.

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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 18 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.

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(continued)

BASES (continued)

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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.

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.

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REFERENCES

1. FSAR, Section [10.3].
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## B 3.7 PLANT SYSTEMS

### B 3.7.5 Emergency Feedwater (EFW) System

#### BASES

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#### 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.

(continued)

## BASES

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### BACKGROUND (continued)

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.

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).

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### 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 the NRC Policy Statement.

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(continued)

BASES (continued)

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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 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.

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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.

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## BASES

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### APPLICABILITY (continued)

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.

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### ACTIONS

#### A.1

With one of the two steam supplies to the turbine driven EFW pump inoperable, action must be taken to restore the steam supply to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

- a. The redundant OPERABLE steam supply to the turbine driven EFW pump(s);
- b. The availability of the redundant OPERABLE motor driven EFW pump; and
- c. The low probability of an event occurring that would require the inoperable steam supply to the turbine driven EFW pump(s).

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.

#### B.1

When one of the required EFW trains (pump or flow path) is inoperable, action must be taken to restore the train to

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## BASES

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### ACTIONS

#### B.1 (continued)

OPERABLE status within 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 AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

#### C.1 and C.2

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.

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BASES

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ACTIONS  
(continued)

D.1

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 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.

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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.

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## BASES

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### SURVEILLANCE REQUIREMENTS

#### SR 3.7.5.1 (continued)

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 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. The [31] day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 3.

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

(continued)

## BASES

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### SURVEILLANCE REQUIREMENTS

#### SR 3.7.5.3 (continued)

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 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 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 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 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.]

<p>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.</p>
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(continued)

## BASES

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### SURVEILLANCE REQUIREMENTS (continued)

#### 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 MODE 5 or 6. 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:

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### REFERENCES

1. FSAR, Section [9.2.7].
  2. FSAR, Section [9.2.8].
  3. ASME, Boiler and Pressure Vessel Code, Section XI.
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## B 3.7 PLANT SYSTEMS

### B 3.7.6 Condensate Storage Tank (CST)

#### BASES

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##### 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).

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##### 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

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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
- 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 Criterion 3 of the NRC Policy Statement.

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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.

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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.

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(continued)

## BASES (continued)

## 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 24 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 verify the OPERABILITY of the backup water supply. 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 timeperiod, 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 [18 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  
REQUIREMENTSSR 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

(continued)



BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.6.1 (continued)

alarms, to alert the operator to abnormal deviations in CST levels.

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REFERENCES

1. FSAR, Section [9.2.6].
  2. FSAR, Chapter [6].
  3. FSAR, Chapter [15].
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## B 3.7 PLANT SYSTEMS

### B 3.7.7 Component Cooling Water (CCW) System

#### BASES

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##### 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.

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##### 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.

(continued)

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## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

The CCW System is designed to perform its function with a single failure of any active component assuming a loss of offsite power.

The CCW System also functions to cool the unit from [DHR] entry conditions ( $T_{\text{cold}} < [350]^{\circ}\text{F}$ ) to MODE 5 ( $T_{\text{cold}} < [200]^{\circ}\text{F}$ ) during normal and post accident operations. The time required to cool from  $[350]^{\circ}\text{F}$  to  $[200]^{\circ}\text{F}$  is a function of the number of CCW and [DHR] trains operating. One CCW train is sufficient to remove decay heat during subsequent operations with  $T_{\text{cold}} < [200]^{\circ}\text{F}$ .

The CCW System satisfies Criterion 3 of the NRC Policy Statement.

---

### 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.

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### APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system that must be prepared to perform its post

(continued)

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## BASES

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APPLICABILITY (continued)      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.

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## 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.

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.

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## 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

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.7.1 (continued)

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 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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.7.3 (continued)

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.

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REFERENCES

1. FSAR, Section [9.2.2].
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## B 3.7 PLANT SYSTEMS

### B 3.7.8 Service Water System (SWS)

#### BASES

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##### 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].

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##### 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

(continued)

## BASES

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### APPLICABLE SAFETY ANALYSES (continued)

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.

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 the NRC Policy Statement.

---

### 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.

(continued)

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BASES

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APPLICABILITY (continued)      In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports.

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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 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

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## BASES

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### SURVEILLANCE REQUIREMENTS

#### SR 3.7.8.1 (continued)

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.

#### 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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.8.3 (continued)

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.

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REFERENCES

1. FSAR, Section [9.2.1].
  2. FSAR, Section [6.2].
  3. FSAR, Section [6.3].
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## B 3.7 PLANT SYSTEMS

### B 3.7.9 Ultimate Heat Sink (UHS)

#### BASES

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##### 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.

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(continued)

BASES (continued)

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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 the NRC Policy Statement.

---

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.

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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.

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(continued)

BASES (continued)

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 inoperable in one or more cooling towers, the number of available systems, and the time required to complete the Required Action.

B.1 and B.2

[If the cooling tower fan cannot be restored to OPERABLE status within the associated Completion Time, or] if the UHS is inoperable [for reasons other than Condition A], the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE  
REQUIREMENTS

SR 3.7.9.1

This SR 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  $\geq [ \quad ]$  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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.9.2 (continued)

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  $\leq [90]^{\circ}\text{F}$ .

SR 3.7.9.3

Operating each cooling tower fan for  $\geq [15]$  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.

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REFERENCES

1. FSAR, Section [9.2.5].
  2. Regulatory Guide 1.27.
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## B 3.7 PLANT SYSTEMS

### B 3.7.10 Control Room Emergency Ventilation System (CREVS)

#### BASES

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##### 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<sup>2</sup> 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.

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##### 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).

(continued)



BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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.

The CREVS satisfies Criterion 3 of the NRC Policy Statement.

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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.

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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.

(continued)

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BASES

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APPLICABILITY (continued) During movement of irradiated fuel assemblies [and during CORE ALTERATIONS], the CREVS must be OPERABLE to cope with a release due to a fuel handling accident.

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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 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, and C.2.2

[In MODE 5 or 6, or] during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS], 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 C.1 is modified by a Note indicating to place the system in the emergency mode if automatic transfer to emergency mode is inoperable.

(continued)

BASES

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ACTIONS

C.1, C.2.1, and C.2.2 (continued)

An alternative to Required Action C.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.

D.1

[In MODE 5 or 6, or] during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS], when two CREVS trains are inoperable, action must be taken immediately to suspend activities that could release radioactivity that could enter 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

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4, 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.

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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  $\geq 10$  continuous hours with the heaters energized. Systems without heaters need only be operated for  $\geq 15$  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.

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

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 CREVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). 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  $\geq$  [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  $\leq$  [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.

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REFERENCES

1. FSAR, Section [9.4].
  2. FSAR, Chapter [15].
  3. Regulatory Guide 1.52.
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## B 3.7 PLANT SYSTEMS

### B 3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

#### BASES

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##### 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).

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##### 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

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

The CREATCS satisfies Criterion 3 of the NRC Policy Statement.

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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.

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APPLICABILITY

In MODES 1, 2, 3, 4, [5, and 6,] and during movement of irradiated fuel assemblies and during CORE ALTERATIONS, the CREATCS must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY requirements following isolation of the control room.

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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.

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BASES

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ACTIONS

A.1 (continued)

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 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 irradiated fuel [, or during CORE ALTERATIONS], 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 irradiated fuel assemblies [, or during CORE ALTERATIONS], 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

(continued)

## BASES

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### ACTIONS

#### D.1 (continued)

[ 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.

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### 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.

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### REFERENCES

1. FSAR, Section [9.4].
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## B 3.7 PLANT SYSTEMS

### B 3.7.12 Emergency Ventilation System (EVS)

#### BASES

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##### 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).

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##### 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.

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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.

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 the NRC Policy Statement.

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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.
- 

APPLICABILITY

In MODES 1, 2, 3, and 4, the EVS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODES 5 and 6, the EVS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

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(continued)

BASES (continued)

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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 and B.2

If the EVS 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.

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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  $\geq 10$  continuous hours with the heaters energized. Systems without heaters need only be operated for  $\geq 15$  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.

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BASES

SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.7.12.2

This SR verified that the required EVS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The EVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). 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 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.

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BASES (continued)

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- 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.
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## B 3.7 PLANT SYSTEMS

### B 3.7.13 Fuel Storage Pool Ventilation System (FSPVS)

#### BASES

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#### 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. 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),

(continued)

BASES

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BACKGROUND (continued)	because it may be used for normal as well as post accident, atmospheric cleanup functions.
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APPLICABLE SAFETY ANALYSES	<p>The FSPVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3, assumes that a certain number of fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident 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).</p> <p>The FSPVS satisfies Criterion 3 of the NRC Policy Statement.</p>
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LCO	<p>[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.</p> <p>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:</p> <ol style="list-style-type: none"><li>1. Fan is OPERABLE;</li><li>2. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and</li><li>3. [Heater, demister,] ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.</li></ol>
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BASES (continued)

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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 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.

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ACTIONS

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 and B.2

In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the associated Completion Time, or when both FSPVS trains are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in 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.

C.1 and C.2

If the inoperable FSPVS train cannot be restored to OPERABLE status within the required Completion Time, during movement

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## BASES

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### ACTIONS

#### C.1 and C.2 (continued)

of irradiated fuel assemblies in the fuel handling area, the OPERABLE FSPVS train must be started immediately or 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 fuel movement, which precludes a fuel handling accident. This action does not preclude the movement of fuel assemblies to a safe position.

#### D.1

When two trains of the FSPVS are inoperable during movement of irradiated fuel assemblies in the fuel handling area, the unit must be placed in a condition in which the LCO does not apply. This LCO involves immediately suspending movement of irradiated fuel assemblies in the fuel handling area. This does not preclude the movement of fuel to a safe position.

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### 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  $\geq 10$  continuous hours with the heaters energized. Systems without heaters need only be operated for  $\geq 15$  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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.13.2 (continued)

Program (VFTP)]. The FSPVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 6). 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 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  $\leq$  [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.

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BASES (continued)

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- 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.
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B 3.7 PLANT SYSTEMS

B 3.7.14 Fuel Storage Pool Water Level

BASES

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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).

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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 Criterion 2 of the NRC Policy Statement.

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(continued)

BASES (continued)

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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.

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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.

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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.

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SURVEILLANCE        SR 3.7.14.1  
REQUIREMENTS

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

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(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.14.1 (continued)

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.

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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.
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B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pool Boron Concentration

BASES

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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  $\geq$  [500] ppm, the criteria that limit the storage of a fuel assembly to specific rack locations are conservatively developed without taking credit for boron.

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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 the NRC Policy Statement.

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LCO

The specified concentration [ $\leq$  [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.

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APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool

(continued)

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BASES

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APPLICABILITY (continued)      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.

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ACTIONS                      A.1, A.2.1, and A.2.2

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.

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.

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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.

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REFERENCES                  None.

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## B 3.7 PLANT SYSTEMS

### B 3.7.16 Spent Fuel Assembly Storage

#### BASES

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##### 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] irradiated 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  $\leq 0.95$  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.

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##### 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 assembly storage satisfies Criterion 2 of the NRC Policy Statement.

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##### 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.

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BASES (continued)

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APPLICABILITY      This LCO applies whenever any fuel assembly is stored in [Region 2] of the spent fuel pool.

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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 to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure [3.7.16-1].

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, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

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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.

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REFERENCES        None.

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## B 3.7 PLANT SYSTEMS

### B 3.7.17 Secondary Specific Activity

#### BASES

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##### 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  $\mu\text{Ci/gm}$  (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). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

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.

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(continued)

BASES (continued)

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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  $0.1 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$ . 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.

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 (ADV). 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 the NRC Policy Statement.

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LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of  $\leq [0.10] \mu\text{Ci/gm DOSE EQUIVALENT I-131}$  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,

(continued)

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BASES

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LCO  
(continued)            to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

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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.

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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.

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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.

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(continued)

BASES (continued)

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- REFERENCES
1. 10 CFR 100.11.
  2. FSAR, Chapter [15].
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## B 3.7 PLANT SYSTEMS

### B 3.7.18 Steam Generator Level

#### BASES

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#### 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

(continued)

BASES

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BACKGROUND  
(continued)

were presented as the amount of mass inventory in each steam generator versus operating range level and steam superheat.

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.

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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.

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BASES

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APPLICABLE SAFETY ANALYSES (continued)      The steam generator level satisfies Criterion 2 of the NRC Policy Statement.

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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) 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.

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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

(continued)

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BASES

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ACTIONS

B.1 (continued)

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.

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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.

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REFERENCES

1. FSAR, Section [15.4.4].
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.1 AC Sources—Operating

#### BASES

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#### BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate(s)) and the onsite standby power sources (Train A and Train B diesel generators (DGs)). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single DG.

Offsite power is supplied to the unit switchyard(s) from the transmission network by [two] transmission lines. From the switchyard(s), two electrically and physically separated circuits provide AC power, through [step down station auxiliary transformers], to the 4.16 kV ESF buses. A detailed description of the offsite power network and the circuits to the Class 1E ESF buses is found in the FSAR, Chapter [8] (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus(es).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System. Within [1 minute] after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the load sequencer.

The onsite standby power source for each 4.16 kV ESF bus is a dedicated DG. DGs [11] and [12] are dedicated to ESF buses [11] and [12], respectively. A DG starts

(continued)

BASES

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BACKGROUND  
(continued)

automatically on a Reactor Coolant System (RCS) pressure signal or on an [ESF bus degraded voltage or undervoltage signal] (refer to LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation" [and LCO 3.3.8, "Emergency Diesel Generator (EDG) Loss of Power Starts (LOPS)"]). After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with a safety injection (SI) signal. The DGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone. Following the trip of offsite power, [a sequencer/an undervoltage signal] strips nonpermanent loads from the ESF bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within [1] minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for Train A and Train B DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is [7000] kW with [10]% overload permissible for up to 2 hours in any 24 hour period. The ESF loads that are powered from the 4.16 kV ESF buses are listed in Reference 2.

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APPLICABLE  
SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the FSAR, Chapter [6] (Ref. 4) and Chapter [15] (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite power or all onsite AC power; and
- b. A worst-case single failure.

The AC sources satisfy Criterion 3 of NRC Policy Statement.

---

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System and separate and independent DGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit.

[ In addition, one required automatic load sequencer per train must be OPERABLE. ]

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses.

[ Offsite circuit #1 consists of Safeguards Transformer B, which is supplied from Switchyard Bus B, and is fed through breaker 52-3 powering the ESF transformer XNB01, which, in turn, powers the #1 ESF bus through its normal feeder breaker. Offsite circuit #2 consists of the Startup Transformer, which is normally fed from the Switchyard Bus A, and is fed through breaker PA 0201 powering the ESF ]

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BASES

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LCO  
(continued)

[ transformer, which, in turn, powers the #2 ESF bus through  
its normal feeder breaker. ]

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This will be accomplished within [10] seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions, such as DG in standby with the engine hot and DG in standby with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode.

Proper sequencing of loads, [including tripping of non-essential loads,] is a required function for DG OPERABILITY.

The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the DGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical. [A circuit may be connected to more than one ESF bus, with fast-transfer capability to the other circuit OPERABLE, and not violate separation criteria. A circuit that is not connected to an ESF bus is required to have OPERABLE fast-transfer interlock mechanisms to at least two ESF buses to support OPERABILITY of that circuit.]

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APPLICABILITY

The AC sources [and sequencers] are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOs or abnormal transients; and

(continued)

BASES

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APPLICABILITY  
(continued)

- b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources—Shutdown."

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ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

Reviewer's Note: The turbine driven auxiliary feedwater pump is only required to be considered a redundant required feature, and, therefore, required to be determined OPERABLE by this Required Action, if the design is such that the remaining OPERABLE motor or turbine driven auxiliary feedwater pump(s) is not by itself capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features. These features are powered from the redundant AC electrical power train. This includes motor driven emergency feedwater pumps. Single train systems, such as turbine driven emergency feedwater pumps, may not be included.

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BASES

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ACTIONS

A.2 (continued)

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

- a. The train has no offsite power supplying it loads; and
- b. A required feature on the other train is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the

(continued)



BASES

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ACTIONS

A.3 (continued)

potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 144 hours, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional 72 hours (for a total of 9 days) allowed prior to complete restoration of the LCO. The 6 day Completion Time provides a limit on the time allowed in a 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 the 72 hour and 6 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action A.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of

(continued)

BASES

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ACTIONS

B.1 (continued)

the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

Reviewer's Note: The turbine driven auxiliary feedwater pump is only required to be considered a redundant required feature, and, therefore, required to be determined OPERABLE by this Required Action, if the design is such that the remaining OPERABLE motor or turbine driven auxiliary feedwater pump(s) is not by itself capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. This includes motor driven emergency feedwater pumps. Single train systems, such as turbine driven emergency feedwater pumps, are not included. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable DG.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

- a. An inoperable DG exists; and
- b. A required feature on the other train is inoperable.

(continued)

BASES

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ACTIONS

B.2 (continued)

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single-failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DG(s). If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DG(s), the other DG(s) would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG(s), performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

(continued)

BASES

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ACTIONS

B.3.1 and B.3.2 (continued)

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the [plant corrective action program] will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), [24] hours is reasonable to confirm that the OPERABLE DG(s) is not affected by the same problem as the inoperable DG.

B.4

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition B for a period that should not exceed 72 hours.

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 144 hours, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 9 days) allowed prior to complete restoration of the LCO. The 6 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Condition A and Condition B are entered concurrently. The "AND" connector between the 72 hour and 6 day Completion Times

(continued)

BASES

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ACTIONS

B.4 (continued)

means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. This includes motor driven auxiliary feedwater pumps. Single train features, such as turbine driven auxiliary pumps, are not included in the list.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

- a. All required offsite circuits are inoperable; and
- b. A required feature is inoperable.

(continued)

BASES

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ACTIONS

C.1 and C.2 (continued)

If at any time during the existence of Condition C (two offsite circuits inoperable) and a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

- a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and
- b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst-case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may

(continued)

BASES

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ACTIONS

C.1 and C.2 (continued)

continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation would continue in accordance with Condition A.

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems—Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 12 hours.

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

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BASES

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ACTIONS  
(continued)

E.1

With Train A and Train B DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Reference 6, with both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

F.1

The sequencer(s) is an essential support system to [both the offsite circuit and the DG associated with a given ESF bus]. [Furthermore, the sequencer is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus.] Therefore, loss of an [ESF bus sequencer] affects every major ESF system in the [division]. The [12] hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer OPERABILITY. This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods when the sequencer is inoperable is minimal.

This Condition is preceded by a Note that allows the Condition to be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads under any conditions. Implicit in this Note is the concept that the Condition must be retained if any sequencer failure mode results in the inability to start all or part of the safety

(continued)



## BASES

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### ACTIONS

#### F.1 (continued)

loads when required, regardless of power availability, or results in overloading the offsite power circuit to a safety bus during an event thereby causing its failure. Also implicit in the Note is that the Condition is not applicable to any train that does not have a sequencer.

#### G.1 and G.2

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

#### H.1

Condition H corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

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### SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 9), and Regulatory Guide 1.137 (Ref. 10), as addressed in the FSAR.

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of [3740] V is 90% of the nominal 4160 V output voltage. This value, which is specified in ANSI C84.1 (Ref. 11), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The specified maximum steady state output voltage of [4756] V is equal to the maximum operating voltage specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to  $\pm 2\%$  of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2 for SR 3.8.1.2) to indicate that all DG starts for these Surveillances may be preceded an

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.7 (continued)

engine prelube period and followed by a warmup period prior to loading by an engine prelube period.

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the DGs are started from standby conditions. Standby conditions for a DG means that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

In order to reduce stress and wear on diesel engines, some manufacturers recommend a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. This is the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer.

SR 3.8.1.7 requires that, at a 184 day Frequency, the DG starts from standby conditions and achieves required voltage and frequency within 10 seconds. The 10 second start requirement supports the assumptions of the design basis LOCA analysis in the FSAR, Chapter [15] (Ref. 5).

The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 3) when a modified start procedure as described above is used. If a modified start is not used, the 10 second start requirement of SR 3.8.1.7 applies.

Since SR 3.8.1.7 requires a 10 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.

The normal 31 day Frequency for SR 3.8.1.2 (see Table 3.8.1-1, "Diesel Generator Test Schedule," in the accompanying LCO) is consistent with Regulatory Guide 1.9 (Ref. 3). The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

(continued)

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between [0.8 lagging] and [1.0]. The [0.8] value is the design rating of the machine, while the [1.0] is an operational limitation [to ensure circulating currents are minimized]. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The normal 31 day Frequency for this Surveillance (Table 3.8.1-1) is consistent with Regulatory Guide 1.9 (Ref. 3).

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit will not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank [and engine mounted tank] is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.4 (continued)

selected to ensure adequate fuel oil for a minimum of 1 hour of DG operation at full load plus 10%.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day [and engine mounted] tanks once every [31] days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 10). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.6 (continued)

The Frequency for this SR is variable, depending on individual system design, with up to a [92] day interval. The [92] day Frequency corresponds to the testing requirements for pumps as contained in the ASME Code, Section XI (Ref. 12); however, the design of fuel transfer systems is such that pumps will operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day [and engine mounted] tanks during or following DG testing. In such a case, a 31 day Frequency is appropriate. Since proper operation of fuel transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs.

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Transfer of each [4.16 kV ESF bus] power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The [18 month] Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Credit may be taken for unplanned events that satisfy this SR.

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. For the CR-3 emergency DGs, the largest single load is 616 kW (HPI pump). After performance of SR 3.8.1.17, the diesel load is reduced to approximately 1200 kW and allowed to run at this load for 3 to 5 minutes. The load is then reduced to  $\geq 616$  kW and the DGs output breaker is opened. Verification that the DG did not trip is made. This Surveillance may be accomplished by:

- a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or
- b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

As required by IEEE-308 (Ref. 13), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The [3] seconds specified is equal to 60% of a typical 5 second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover to following load rejection. The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9).

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.9 (continued)

This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Credit may be taken for unplanned events that satisfy this SR. In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, Note 2 requires that, if synchronized to offsite power, testing must be performed using a power factor  $\leq [0.9]$ . This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

- a. Performance of the SR will not render any safety system or component inoperable;
- b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and
- c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

SR 3.8.1.10

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG will not trip upon loss of the load. These

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REQUIREMENTS

SR 3.8.1.10 (continued)

acceptance criteria provide DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor  $\leq [0.9]$ . This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9) and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbation to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems.

Credit may be taken for unplanned events that satisfy this SR.

Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

- a. Performance of the SR will not render any safety system or component inoperable;
- b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and
- c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

(continued)

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.8.1.11

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the non-essential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG auto-start time of [10] seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads can not actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, high pressure injection systems are not capable of being operated at full flow, or decay heat removal (DHR) systems performing a DHR function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.11 (continued)

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time ([10] seconds) from the design basis actuation signal (LOCA signal) and operates for  $\geq 5$  minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.12.d and SR 3.8.1.12.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power.

The requirement to verify the connection of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads can not actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, high pressure injection systems are not capable of being operated at full flow, or DHR systems performing a DHR function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.12 (continued)

The Frequency of [18 months] takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.13

This Surveillance demonstrates that DG noncritical protective functions (e.g., high jacket water temperature) are bypassed on a loss of voltage signal concurrent with an ESF actuation test signal, and critical protective functions (engine overspeed, generator differential current[, low lube oil pressure, high crankcase pressure, and start failure relay]) trip the DG to avert substantial damage to the DG unit. The noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The [18 month] Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has

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SR 3.8.1.13 (continued)

shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service. Credit may be taken for unplanned events that satisfy this SR.

Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

- a. Performance of the SR will not render any safety system or component inoperable;
- b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and
- c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

SR 3.8.1.14

Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3), requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours,  $\geq$  [2] hours of which is at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

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SR 3.8.1.14 (continued)

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing must be performed using a power factor of  $\leq [0.9]$ . This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections, in accordance with vendor recommendations, in order to maintain DG OPERABILITY.

The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 7), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within [10 seconds]. The [10 second] time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(5).

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.15 (continued)

inspections, in accordance with vendor recommendations, in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least [2] hours at full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.16

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and the DG can be returned to ready to load status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open and can receive and auto-close signal on bus undervoltage, and the load sequence timers are reset.

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to ready to load operation if a LOCA actuation signal is received during operation in the test

(continued)

BASES

SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.17 (continued)

mode. Ready to load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 13), paragraph 6.2.6(2).

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(8), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.18

Under accident [and loss of offsite power] conditions loads are sequentially connected to the bus by the [automatic load sequencer]. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The [10]% load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

(continued)



BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.18 (continued)

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

- a. Performance of the SR will not render any safety system or component inoperable;
- b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and
- c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

SR 3.8.1.19

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.19 (continued)

functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of [18 months] takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of [18 months].

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.20

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9).

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated, and temperature maintained consistent with manufacturer recommendations.

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

Diesel Generator Test Schedule

The DG test schedule (Table 3.8.1-1) implements the recommendations of Revision 3 to Regulatory Guide 1.9 (Ref. 3). The purpose of this test schedule is to provide timely test data to establish a confidence level associated with the goal to maintain DG reliability above 0.95 per demand.

According to Regulatory Guide 1.9, Revision 3 (Ref. 3), each DG should be tested at least once every 31 days. Whenever a DG has experienced 4 or more valid failures in the last 25 valid tests, the maximum time between tests is reduced to 7 days. Four failures in 25 valid tests is a failure rate of 0.16, or the threshold of acceptable DG performance, and hence may be an early indication of the degradation of DG reliability. When considered in the light of a long history of tests, however, 4 failures in the last 25 valid tests may only be a statistically probable distribution of random events. Increasing the test Frequency will allow for a more timely accumulation of additional test data upon which to base judgment of the reliability of the DG. The increased test Frequency must be maintained until seven consecutive, failure free tests have been performed.

The Frequency for accelerated testing is 7 days, but no less than 24 hours. Tests conducted at intervals of less than 24 hours may be credited for compliance with Required Actions. However, for the purpose of re-establishing the normal 31-day Frequency, a successful test at an interval of less than 24 hours should be considered an invalid test and not count towards the seven consecutive failure free starts, and the consecutive test count is not reset.

A test interval in excess of 7 days (or 31 days, as appropriate) constitutes a failure to meet the SRs and results in the associated DG being declared inoperable. It does not, however, constitute a valid test or failure of the DG, and any consecutive test count is not reset.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter [8].

(continued)

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BASES

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REFERENCES  
(continued)

3. Regulatory Guide 1.9, Rev. [3], [date].
  4. FSAR, Chapter [6].
  5. FSAR, Chapter [15].
  6. Regulatory Guide 1.93, Rev. [0], [date].
  7. Generic Letter 84-15.
  8. 10 CFR 50, Appendix A, GDC 18.
  9. Regulatory Guide 1.108, Rev. [1], [August 1977].
  10. Regulatory Guide 1.137, Rev. [ ], [date].
  11. ANSI C84.1-1982.
  12. ASME, Boiler and Pressure Vessel Code, Section XI.
  13. IEEE Standard 308-[1978].
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.2 AC Sources—Shutdown

#### BASES

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BACKGROUND	A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources—Operating."
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APPLICABLE SAFETY ANALYSES	The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:
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- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst-case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4 various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown MODES based on:

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration;
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both;
- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems; and
- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of the NRC Policy Statement.

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LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems—Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required

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BASES

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LCO  
(continued)

offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es). Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit.

Offsite circuit #1 consists of Safeguards Transformer B, which is supplied from Switchyard Bus B, and is fed through breaker 52-3 powering the ESF transformer XNB01, which, in turn, powers the #1 ESF bus through its normal feeder breaker. The second offsite circuit consists of the Startup Transformer, which is normally fed from the Switchyard Bus A, and is fed through breaker PA 0201 powering the ESF transformer, which, in turn, powers the #2 ESF bus through its normal feeder breaker.

The DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This sequence must be accomplished within [10] seconds. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby at ambient conditions.

Proper sequencing of loads, including tripping of non-essential loads, is a required function for DG OPERABILITY.

In addition, proper sequencer operation is an integral part of offsite circuit OPERABILITY since its inoperability impacts on the ability to start and maintain energized loads required OPERABLE by LCO 3.8.10.

It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains.

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(continued)

BASES (continued)

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- APPLICABILITY      The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:
- a.    Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;
  - b.    Systems needed to mitigate a fuel handling accident are available;
  - c.    Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
  - d.    Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.
- The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.
- 

ACTIONS

A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF train. Although two trains are required by LCO 3.8.10, the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By the allowance of the option to declare features inoperable with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC

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BASES

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ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4  
(continued)

power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SDM is maintained.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS are not entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 provides the appropriate restrictions for the situation involving a de-energized train.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.2.1 (continued)

to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.6 is not required to be met because the required OPERABLE DG(s) is not required to undergo periods of being synchronized to the offsite circuit. SR 3.8.1.9 is excepted because starting independence is not required with the DG(s) that is not required to be OPERABLE.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 4160 V ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

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REFERENCES

None.

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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

#### BASES

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##### BACKGROUND

Each diesel generator (DG) is provided with a storage tank having a fuel oil capacity sufficient to operate that diesel for a period of 7 days while the DG is supplying maximum post loss of coolant accident load demand discussed in the FSAR, Section [9.5.4.2] (Ref. 1). The maximum load demand is calculated using the assumption that at least two DGs are available. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from storage tank to day tank by either of two transfer pumps associated with each storage tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve or tank to result in the loss of more than one DG. All outside tanks, pumps, and piping are located underground.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting a minimum of [7] days of operation. [The onsite storage in addition to the engine oil sump is sufficient to ensure 7 days of continuous operation.] This supply is sufficient to allow the operator to replenish lube oil from outside sources.

Each DG has an air start system with adequate capacity for five successive start attempts on the DG without recharging the air start receiver(s).

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BASES (continued)

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APPLICABLE  
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 4) and Chapter [15] (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, and the air start subsystem support the operation of the standby AC power sources, they satisfy Criterion 3 of the NRC Policy Statement.

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LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of full load operation. It is also required to meet specific standards for quality. Additionally, sufficient lube oil supply must be available to ensure the capability to operate at full load for 7 days. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources" Operating," and LCO 3.8.2, "AC Sources" Shutdown."

The starting air system is required to have a minimum capacity for five successive DG start attempts without recharging the air start receivers.

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APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil, lube oil, and the starting air subsystem support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil,

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BASES

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APPLICABILITY      and starting air are required to be within limits when the  
(continued)      associated DG is required to be OPERABLE.

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ACTIONS      The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

A.1

In this Condition, the 7 day fuel oil supply for a DG is not available. However, the Condition is restricted to fuel oil level reductions, that maintain at least a 6 day supply. These circumstances may be caused by events, such as full load operation required after an inadvertent start while at minimum required level; or feed and bleed operations, which may be necessitated by increasing particulate levels or any number of other oil quality degradations. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

With lube oil inventory < 500 gal, sufficient lube oil to support 7 days of continuous DG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration

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BASES

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ACTIONS

B.1 (continued)

of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.5. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.4 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

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BASES

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ACTIONS  
(continued)

E.1

With starting air receiver pressure < [225] psig, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver pressure is > [125] psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With a Required Action and associated Completion Time not met, or one or more DGs with fuel oil, lube oil, or starting air subsystem not within limits for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The 7 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.2

This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of full load

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.3.2 (continued)

operation for each DG. The [500] gal requirement is based on the DG manufacturer consumption values for the run time of the DG. Implicit in this SR is the requirement to verify the capability to transfer the lube oil from its storage location to the DG, when the DG lube oil sump does not hold adequate inventory for 7 days of full load operation without the level reaching the manufacturer recommended minimum level.

A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the unit staff.

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

- a. Sample the new fuel oil in accordance with ASTM D4057-[ ] (Ref. 6);
- b. Verify in accordance with the tests specified in ASTM D975-[ ] (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of  $\geq 0.83$  and  $\leq 0.89$  or an API gravity at 60°F of  $\geq 27^\circ$  and  $\leq 39^\circ$ , a kinematic viscosity at 40°C of  $\geq 1.9$  centistokes and  $\leq 4.1$  centistokes, and a flash point of  $\geq 125^\circ\text{F}$ ; and
- c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-[ ] (Ref. 6).

(continued)



BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.3.3 (continued)

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-[ ] (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975-[ ] (Ref. 6), except that the analysis for sulfur may be performed in accordance with ASTM D1552-[ ] (Ref. 6) or ASTM D2622-[ ] (Ref. 6). The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-[ ], Method A (Ref. 6). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. [For those designs in which the total stored fuel oil volume is contained in two or more interconnected tanks, each tank must be considered and tested separately.]

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of [five] engine start cycles without recharging. [A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed.] The pressure specified in this SR is intended to reflect the lowest value at which the [five] starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every [31] days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

SR 3.8.3.6

Draining of the fuel oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning are required at

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.3.6 (continued)

10 year intervals by Regulatory Guide 1.137 (Ref. 2), paragraph 2.f. This SR also requires the performance of the ASME Code, Section XI (Ref. 8), examinations of the tanks. To preclude the introduction of surfactants in the fuel oil system, the cleaning should be accomplished using sodium hypochlorite solutions, or their equivalent, rather than soap or detergents. This SR is for preventive maintenance. The presence of sediment does not necessarily represent a failure of this SR, provided that accumulated sediment is removed during performance of the Surveillance.

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REFERENCES

1. FSAR, Section [9.5.4.2].
  2. Regulatory Guide 1.137.
  3. ANSI N195-1976, Appendix B.
  4. FSAR, Chapter [6].
  5. FSAR, Chapter [15].
  6. ASTM Standards: D4057-[ ]; D975-[ ];  
D4176-[ ]; D1552-[ ]; D2622-[ ];  
D2276-[ ], Method A.
  7. ASTM Standards, D975, Table 1.
  8. ASME, Boiler and Pressure Vessel Code, Section XI.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.4 DC Sources—Operating

#### BASES

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##### BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC vital bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The [125/250] VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems ([Train A and Train B]). Each subsystem consists of [two] 125 VDC batteries [(each battery [50]% capacity)], the associated battery charger[s] for each battery, and all the associated control equipment and interconnecting cabling.

The 250 VDC source is obtained by use of the two 125 VDC batteries connected in series. Additionally, there is [one] spare battery charger per subsystem, which provides backup service in the event that the preferred battery charger is out of service. If the spare battery charger is substituted for one of the preferred battery chargers, then the requirements of independence and redundancy between subsystems are maintained.

During normal operation, the [125/250] VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

The [Train A and Train B] DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, [4.16] kV switchgear, and [480] V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the AC vital buses.

(continued)

## BASES

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### BACKGROUND (continued)

The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distributions System—Operating," and for LCO 3.8.10, "Distribution Systems—Shutdown."

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours and to perform three complete cycles of intermittent loads discussed in the FSAR, Chapter [8] (Ref. 4).

Each 125/250 VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.

The batteries for Train A and Train B DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. Battery size is based on 125% of required capacity and, after selection of an available commercial battery, results in a battery capacity in excess of 150% of required capacity. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery discussed in the FSAR, Chapter [8] (Ref. 4). The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

Each Train A and Train B DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads discussed in the FSAR, Chapter [8] (Ref. 4).

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### APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 6) and Chapter [15] (Ref. 7), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite AC power or all onsite AC power; and
- b. A worst-case single failure.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

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LCO

The DC electrical power subsystems, each subsystem consisting of [two] batteries, battery charger [for each battery] and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any train DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires all required batteries and respective chargers to be operating and connected to the associated DC bus(es).

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APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

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BASES

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APPLICABILITY  
(continued)

- b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.5, "DC Sources—Shutdown."

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ACTIONS

A.1

Condition A represents one train with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected train. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution system train.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst-case single failure would, however, result in the complete loss of the remaining 250/125 VDC electrical power subsystems with attendant loss of ESF functions, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 8) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

B.1 and B.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to 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

(continued)

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BASES

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ACTIONS

B.1 and B.2 (continued)

within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 8).

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 9).

SR 3.8.4.2

Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each intercell, interrack, intertier, and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The limits established for this SR must be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.

The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The 12 month Frequency for this SR is consistent with IEEE-450 (Ref. 9), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of intercell, interrack, intertier, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.4.

Reviewer's Note: The requirement to verify that terminal connections are clean and tight applies only to nickel cadmium batteries as per IEEE Standard P1106, "IEEE Recommended Practice for Installation, Maintenance, Testing and Replacement of Vented Nickel - Cadmium Batteries for Stationary Applications." This requirement may be removed for lead acid batteries.

The connection resistance limits for SR 3.8.4.5 shall be no more than 20% above the resistance as measured during installation, or not above the ceiling value established by the manufacturer.

The Surveillance Frequencies of [12] months is consistent with IEEE-450 (Ref. 9), which recommends cell to cell and terminal connection resistance measurement on a yearly basis.

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.8.4.6

This SR requires that each battery charger be capable of supplying [400] amps and [250/125] V for  $\geq$  [8] hours. These requirements are based on the design capacity of the chargers (Ref. 4). According to Regulatory Guide 1.32 (Ref. 10), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensure that these requirements can be satisfied.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these [18 month] intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.4.7

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 10) and Regulatory Guide 1.129 (Ref. 11), which state that the battery service test should be performed during refueling operations, or at some other outage, with intervals between tests not to exceed [18 months].

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test once per 60 months.

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.4.7 (continued)

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

A battery modified performance discharge test is described in the Bases for SR 3.8.4.7. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.4.8 (continued)

SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 9) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity  $\geq$  100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 9), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is  $\geq$  [10%] below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 9).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

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REFERENCES

1. 10 CFR.50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-[1978].
4. FSAR, Chapter [8].
5. IEEE-485-[1983], June 1983.

(continued)

BASES

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REFERENCES  
(continued)

6. FSAR, Chapter [6].
  7. FSAR, Chapter [15].
  8. Regulatory Guide 1.93, December 1974.
  9. IEEE-450-[1987].
  10. Regulatory Guide 1.32, February 1977.
  11. Regulatory Guide 1.129, December 1974.
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.5 DC Sources—Shutdown

#### BASES

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**BACKGROUND** A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources—Operating."

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**APPLICABLE SAFETY ANALYSES** The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [14] (Ref. 2), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

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**LCO** The DC electrical power subsystems, each subsystem consisting of two batteries, one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the train, are required to be

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BASES

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LCO  
(continued)      OPERABLE to support required trains of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems—Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

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APPLICABILITY      The DC electrical power sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies, provide assurance that:

- a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;
- b. Required features needed to mitigate a fuel handling accident are available;
- c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

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ACTIONS      A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, the remaining train with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances this option may involve undesired administrative efforts. Therefore, the

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BASES

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ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.8. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

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BASES (continued)

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- REFERENCES
1. FSAR, Chapter [6].
  2. FSAR, Chapter [14].
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.6 Battery Cell Parameters

#### BASES

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**BACKGROUND** This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4, "DC Sources—Operating," and LCO 3.8.5, "DC Sources—Shutdown."

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**APPLICABLE SAFETY ANALYSES** The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [15] (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least one train of DC sources OPERABLE during accident conditions, in the event of:

- a. An assumed loss of all offsite AC power or all onsite AC power; and
- b. A worst-case single failure.

Battery cell parameters satisfy Criterion 3 of the NRC Policy Statement.

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**LCO** Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Electrolyte limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met.

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BASES (continued)

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APPLICABILITY      The battery cell parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery electrolyte is only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

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ACTIONS            A.1, A.2, and A.3

With one or more cells in one or more batteries not within limits (i.e., Category A limits not met or Category B limits not met or Category A and B limits not met) but within the Category C limits specified in Table 3.8.6-1 in the accompanying LCO, the battery is degraded but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cells. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery will still be capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because specific gravity measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A and B limits. This periodic verification is consistent with the normal Frequency of pilot cell Surveillances.

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BASES

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ACTIONS

A.1, A.2, and A.3 (continued)

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below 60°F, are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of pilot cells.

SR 3.8.6.2

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 3). In addition, within 24 hours of a battery discharge < [110] V or a battery overcharge > [150] V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ [110] V, do not constitute a battery discharge

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.6.2 (continued)

provided battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells is  $> [60]^{\circ}\text{F}$  is consistent with a recommendation of IEEE-450 (Ref. 3), which states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 3), with the extra  $\frac{1}{4}$  inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote a to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates

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BASES

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SURVEILLANCE  
REQUIREMENTS

Table 3.8.6-1 (continued)

suffer no physical damage and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is  $\geq 2.13$  V per cell. This value is based on a recommendation of IEEE-450 (Ref. 3), which states that prolonged operation of cells  $< 2.13$  V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is  $\geq [1.200]$  (0.015 below the manufacturer fully charged nominal specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 3), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is  $\geq [1.195]$  (0.020 below the manufacturer fully charged, nominal specific gravity) with the average of all connected cells  $> [1.205]$  (0.010 below the manufacturer fully charged, nominal specific gravity). These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

Table 3.8.6-1 (continued)

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limits for float voltage is based on IEEE-450 (Ref. 3), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limits of average specific gravity  $\geq [1.195]$  is based on manufacturer recommendations (0.020 below the manufacturer recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

The footnotes to Table 3.8.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is  $< [2]$  amps on float charge. This current provides, in general, an indication of overall battery condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 3). Footnote (c) to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

Table 3.8.6-1 (continued)

up to [7] days following a battery recharge. Within [7] days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than [7] days.

Reviewer's Note: The value of [2] amps used in footnote (b) and (c) is the nominal value for float current established by the battery vendor as representing a fully charged battery with an allowance for overall battery condition.

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REFERENCES

1. FSAR, Chapter [6].
  2. FSAR, Chapter [15].
  3. IEEE-450-[1980].
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.7 Inverters—Operating

#### BASES

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##### BACKGROUND

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital bus. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station battery provides an uninterruptible power source for the instrumentation and controls for the Reactor Protection System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in FSAR, Chapter [8] (Ref. 1).

---

##### APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 2), and Chapter [14] (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and
- b. A worst-case single failure.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of the NRC Policy Statement.

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(continued)

BASES (continued)

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LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The four inverters [(two per train)] ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized.

OPERABLE inverters require the associated vital bus to be powered by the inverter with output voltage and frequency within tolerances, and power input to the inverter from a [125 VDC] station battery. Alternatively, power supply may be from an internal AC source via rectifier as long as the station battery is available as the uninterruptible power supply.

This LCO is modified by a Note that allows [one/two] inverters to be disconnected from a [common] battery for  $\leq 24$  hours, if the vital bus(es) is powered from a [Class 1E constant voltage transformer or inverter using internal AC source] during the period and all other inverters are operable. This allows an equalizing charge to be placed on one battery. If the inverters were not disconnected, the resulting voltage condition might damage the inverter[s]. These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 24 hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected AC vital bus while taking into consideration the time required to perform an equalizing charge on the battery bank.

The intent of this Note is to limit the number of inverters that may be disconnected. Only those inverters associated with the single battery undergoing an equalizing charge may be disconnected. All other inverters must be aligned to their associated batteries, regardless of the number of inverters or unit design.

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(continued)

BASES (continued)

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- APPLICABILITY      The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:
- a.    Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
  - b.    Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.
- Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters—Shutdown."
- 

ACTIONS

A.1

With a required inverter inoperable, its associated AC vital bus becomes inoperable until it is [manually] re-energized from its [Class 1E constant voltage source transformer or inverter using internal AC source].

For this reason, a Note has been included in Condition A requiring entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems—Operating." This ensures the vital bus is re-energized within 2 hours. Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its constant voltage source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the

(continued)

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## BASES

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### ACTIONS

#### B.1 and B.2 (continued)

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

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### SURVEILLANCE REQUIREMENTS

#### SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

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### REFERENCES

1. FSAR, Chapter [8].
  2. FSAR, Chapter [6].
  3. FSAR, Chapter [14].
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.8 Inverters—Shutdown

#### BASES

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BACKGROUND            A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters—Operating."

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APPLICABLE  
SAFETY ANALYSES      The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [14] (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protection System and Engineered Safety Features Actuation System (ESFAS) instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each AC vital bus during MODES 5 and 6 ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of the NRC Policy Statement.

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(continued)

BASES (continued)

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LCO                    The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters requires that the vital bus be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

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APPLICABILITY        The inverters required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that:

- a.    Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;
- b.    Systems needed to mitigate a fuel handling accident are available;
- c.    Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- d.    Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

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ACTIONS                A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, "Distribution Systems—Shutdown," the remaining OPERABLE inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity

(continued)

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BASES

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ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions).

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

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(continued)

BASES (continued)

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- REFERENCES
1. FSAR, Chapter [6].
  2. FSAR, Chapter [14].
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## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.9 Distribution Systems—Operating

#### BASES

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##### BACKGROUND

The onsite Class 1E AC, DC, and AC vital bus electrical power distribution systems are divided by train into [two] redundant and independent AC, DC, and AC vital bus electrical power distribution subsystems.

The AC electrical power subsystem for each train consists of a primary Engineered Safety Feature (ESF) 4.16 kV bus and secondary [480 and 120] V buses, distribution panels, motor control centers and load centers. Each [4.16 kV ESF bus] has at least [one separate and independent offsite source of power] as well as a dedicated onsite diesel generator (DG) source. Each [4.16 kV ESF bus] is normally connected to a preferred offsite source. After a loss of the preferred offsite power source to a 4.16 kV ESF bus, a transfer to the alternate offsite source is accomplished by utilizing a time delayed bus undervoltage relay. If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ESF bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries. Additional description of this system may be found in the Bases for LCO 3.8.1, "AC Sources—Operating," and the Bases for LCO 3.8.4, "DC Sources—Operating."

The secondary AC electrical power distribution system for each train includes the safety related load centers, motor control centers, and distribution panels shown in Table B 3.8.9-1.

The 120 VAC vital buses are arranged in two load groups per train and are normally powered from the inverters. The alternate power supply for the vital buses are Class 1E constant voltage source transformers powered from the same train as the associated inverter, and its use is governed by LCO 3.8.7, "Inverters—Operating." Each constant voltage source transformer is powered from a Class 1E AC bus.

There are two independent 125/250 VDC electrical power distribution subsystems (one for each train).

The list of all required distribution buses is presented in Table B 3.8.9-1.

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(continued)

BASES (continued)

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APPLICABLE  
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [14] (Ref. 2), assume ESF systems are OPERABLE. The AC, DC, and AC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC, DC, and AC vital bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite power or all onsite AC electrical power; and
- b. A worst-case single failure.

The distribution systems satisfy Criterion 3 of the NRC Policy Statement.

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LCO

The required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of AC, DC, and AC vital bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The AC, DC, and AC vital bus electrical power distribution subsystems are required to be OPERABLE.

Maintaining the Train A and Train B AC, DC, and AC vital bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

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BASES

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LCO  
(continued)

OPERABLE AC electrical power distribution subsystems require the associated buses, load centers, motor control centers, and distribution panels to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. OPERABLE vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated [inverter via inverted DC voltage, inverter using internal AC source, or Class 1E constant voltage transformer].

In addition, tie breakers between redundant safety related AC, DC, and AC vital bus power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, that could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 4.16 kV buses from being powered from the same offsite circuit.

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APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.10, "Distribution Systems—Shutdown."

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(continued)

BASES (continued)

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ACTIONS

A.1

With one or more required AC electrical power distribution subsystems inoperable, the remaining AC electrical power distribution subsystem in the other train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels must be restored to OPERABLE status within 8 hours.

Condition A worst scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

- a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train to the actions associated with taking the unit to shutdown within this time limit; and
- b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 2 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again

(continued)

BASES

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ACTIONS

A.1 (continued)

become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With one [required] AC vital bus inoperable, the remaining OPERABLE AC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the [required] AC vital bus must be restored to OPERABLE status within 2 hours by powering the bus from the associated [inverter via inverted DC, inverter using internal AC Source, or Class 1E constant voltage transformer].

Condition B represents one AC vital bus without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining vital buses and restoring power to the affected vital bus.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

(continued)

BASES

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ACTIONS

B.1 (continued)

- a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the vital bus distribution system. At this time, an AC train could again become inoperable, and vital bus distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

(continued)

BASES

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ACTIONS  
(continued)

C.1

With DC bus(es) in one train inoperable, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the [required] DC buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

Condition C represents one train without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

- a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected train; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).

(continued)

BASES

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ACTIONS

C.1 (continued)

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable and DC distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

D.1 and D.2

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

E.1

Condition E corresponds to a level of degradation in the electrical distribution system that causes a required safety function to be lost. When more than one Condition is entered, and this results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for

(continued)



BASES

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ACTIONS

E.1 (continued)

continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the [required] AC, DC, and AC vital bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

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REFERENCES

1. FSAR, Chapter [6].
  2. FSAR, Chapter [14].
  3. Regulatory Guide 1.93, December 1974.
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Table B 3.8.9-1 (page 1 of 1)  
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	TRAIN A*	TRAIN B*
AC safety buses	[4160 V]	[ESF Bus] [NB01]	[ESF Bus] [NB02]
	[480 V]	Load Centers [NG01, NG03]	Load Centers [NG02, NG04]
	[480 V]	Motor Control Centers [NG01A, NG01I, NG01B, NG03C, NG03I, NG03D]	Motor Control Centers [NG02A, NG02I, NG02B, NG04C, NG04I, NG04D]
	[120 V]	Distribution Panels [NP01, NP03]	Distribution Panels [NP02, NP04]
DC buses	[125 V]	Bus [NK01]	Bus [NK02]
		Bus [NK03]	Bus [NK04]
		Distribution Panels [NK41, NK43, NK51]	Distribution Panels [NK42, NK44, NK52]
AC vital buses	[120 V]	Bus [NN01]	Bus [NN02]
		Bus [NN03]	Bus [NN04]

\* Each train of the AC and DC electrical power distribution systems is a subsystem.

## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.10 Distribution Systems—Shutdown

#### BASES

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**BACKGROUND** A description of the AC, DC and AC vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems—Operating."

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**APPLICABLE SAFETY ANALYSES** The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [14] (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and AC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC vital bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC, DC, and AC vital bus electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

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(continued)

BASES (continued)

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LCO                      Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components" all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

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APPLICABILITY        The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

- a.    Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;
- b.    Systems needed to mitigate a fuel handling accident are available;
- c.    Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- d.    Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC, DC, and AC vital bus electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

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(continued)

BASES (continued)

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ACTIONS

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystems LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions).

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required decay heat removal (DHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.5 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the DHR ACTIONS would not be entered. Therefore, Required Action A.2.6 is provided to direct declaring DHR inoperable, which results in taking the appropriate DHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the AC, DC, and AC vital bus electrical power distribution subsystems are functioning properly, with all the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

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REFERENCES

1. FSAR, Chapter [6].
  2. FSAR, Chapter [14].
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## B 3.9 REFUELING OPERATIONS

### B 3.9.1 Boron Concentration

#### BASES

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##### BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Unit procedures ensure the specified boron concentration in order to maintain an overall core reactivity of  $k_{\text{eff}} \leq 0.95$  during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical Addition System serves as the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the borated water storage tank into the open reactor vessel by gravity feeding or by the use of the Decay Heat Removal (DHR) System pumps.

The pumping action of the DHR System in the RCS, and the natural circulation due to thermal driving heads in the reactor vessel and the refueling cavity, mix the added concentrated boric acid with the water in the refueling canal. The DHR System is in operation during refueling (see

(continued)

BASES

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BACKGROUND  
(continued)

LCO 3.9.4, "DHR and Coolant Circulation" High Water Level," and LCO 3.9.5, "DHR and Coolant Circulation" Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

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APPLICABLE  
SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the  $k_{eff}$  of the core will remain  $\leq 0.95$  during the refueling operation. Hence, at least a 5% Dk/k margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.

---

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures a core  $k_{eff}$  of  $\leq 0.95$  is maintained during fuel handling operations.

This LCO also requires that coolant be circulated during any boron dilution. Providing forced coolant circulation during changes in boron concentration ensures mixing of the coolant, eliminating the potential for pockets of diluted, unmixed coolant, which may cause loss of required SDM.

(continued)

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BASES

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LCO  
(continued)

Adequate mixing prevents stratification to ensure that dilution induced reactivity changes are gradual, as well as recognizable and controllable by the operator. Forced circulation will also ensure that the boron concentration determined by chemical analysis is representative of the entire coolant volume.

Violation of the LCO could lead to an inadvertent criticality during MODE 6.

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APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a  $k_{eff} \leq 0.95$ . Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," and LCO 3.1.2, "Reactivity Balance," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

---

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, action to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, there is no unique Design Basis

(continued)

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BASES

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ACTIONS

A.3 (continued)

Event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.2 Nuclear Instrumentation

#### BASES

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##### BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors is permitted, provided the LCO requirements are met.

The installed source range neutron flux monitors are BF<sub>3</sub> detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps) with a [5]% instrument accuracy. The detectors also provide continuous visual indication in the control room and an audible alarm to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1. If used, portable detectors should be functionally equivalent to the installed NIS source range monitors.

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##### APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity, such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that the normally available SDM would not be lost, and there is sufficient time for the operator to take corrective action.

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement.

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(continued)

BASES (continued)

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LCO                      This LCO requires two source range neutron flux monitors OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

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APPLICABILITY        In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.9, "Source Range Neutron Flux."

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ACTIONS                A.1 and A.2

With only one [required] source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no [required] source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no [required] source range neutron flux monitor OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

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BASES

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ACTIONS

B.2 (continued)

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.9.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every [18] months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range nuclear is a complete check and re-adjustment of the channels, from the pre-amplifier input to the indicators. The 18 month Frequency is based on the need to perform this Surveillance during the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the [18] month Frequency.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
  2. FSAR, Section [ ].
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## B 3.9 REFUELING OPERATIONS

### B 3.9.3 Containment Penetrations

#### BASES

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##### BACKGROUND

During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown

(continued)

BASES

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BACKGROUND  
(continued)

when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a [42] inch purge penetration and a [42] inch exhaust penetration. The second subsystem, or minipurge system, includes an [8] inch purge penetration and an [8] inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and exhaust penetrations are secured in the closed position. The two valves in each of the two minipurge penetrations can be opened intermittently but are closed automatically by the Engineered Safety Feature Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchangers are necessary to conduct refueling operations. The normal [42] inch purge system is used for this purpose, and all four valves are closed on a reactor building (RB) high radiation signal in accordance with LCO 3.3.15, "Reactor Building (RB) Purge Isolation" High Radiation.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements (Ref. 1).

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(continued)

BASES (continued)

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APPLICABLE SAFETY ANALYSES      During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Canal Water Level," and the minimum decay time of [100] hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in 10 CFR 100. The acceptance limits for offsite radiation exposure are contained in Reference 2.

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.

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LCO      This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity from containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the RB purge isolation signal. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and therefore meet the assumptions used in the safety analysis to ensure releases through the valves are terminated such that radiological doses are within the acceptance limit.

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APPLICABILITY      The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment

(continued)

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BASES

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APPLICABILITY  
(continued)      are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

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ACTIONS              A.1 and A.2

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude moving a component to a safe position.

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SURVEILLANCE  
REQUIREMENTS      SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE automatic RB purge isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO.

As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.3.1 (continued)

radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.15, "RB Purge Isolation" High Radiation," the isolation instrumentation requires a CHANNEL CHECK every 12 hours and a CHANNEL FUNCTIONAL TEST every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

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REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.
  2. FSAR, Section [ ].
  3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.4 Decay Heat Removal (DHR) and Coolant Circulation—High Water Level

#### BASES

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**BACKGROUND** The purposes of the DHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System via the DHR heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the DHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the DHR System.

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**APPLICABLE SAFETY ANALYSES** If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction in boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the DHR System is required to be operational in MODE 6, with the water level  $\geq$  23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the DHR pump for short durations under the condition that the boron concentration is not diluted. This conditional de-energizing of the DHR pump does not result in a challenge to the fission product barrier.

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

Although the DHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the DHR System is retained as a Specification.

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LCO

Only one DHR loop is required for decay heat removal in MODE 6, with a water level  $\geq$  23 ft above the top of the reactor vessel flange. Only one DHR loop is required to be OPERABLE because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one DHR loop must be OPERABLE and in operation to provide:

- a. Removal of decay heat;
- b. Mixing of borated coolant to minimize the possibility of criticality; and
- c. Indication of reactor coolant temperature.

An OPERABLE DHR loop includes a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

Additionally, each DHR loop is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation of one subsystem can maintain the reactor coolant temperature as required.

The LCO is modified by a Note that allows the required DHR loop to be removed from operation for up to 1 hour in an 8 hour period, provided no operation that would cause reduction of the RCS boron concentration is in progress. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to DHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

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(continued)

BASES (continued)

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APPLICABILITY      One DHR loop must be OPERABLE and in operation in MODE 6, with the water level  $\geq$  23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Canal Water Level." Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6, with the water level  $<$  23 ft above the top of the reactor vessel flange, are located in LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level."

---

ACTIONS            DHR loop requirements are met by having one DHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If DHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by adding water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

A.2

If DHR loop requirements are not met, actions shall be taken immediately to suspend the loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is prudent under this condition.

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BASES

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ACTIONS  
(continued)

A.3

If DHR loop requirements are not met, actions shall be initiated immediately in order to satisfy DHR loop requirements.

A.4

If DHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to outside atmosphere shall be closed within 4 hours.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the DHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the DHR System.

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REFERENCES

1. FSAR, Section [ ].
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## B 3.9 REFUELING OPERATIONS

### B 3.9.5 Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level

#### BASES

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##### BACKGROUND

The purposes of the DHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System via the DHR heat exchanger. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the DHR System for normal cooldown/decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the DHR System.

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##### APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the DHR System are required to be OPERABLE, and one is required to be in operation, to prevent this challenge.

Although the DHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

reduction. Therefore, the DHR System is retained as a Specification.

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LCO

In MODE 6, with the water level  $< 23$  ft above the top of the reactor vessel flange, two DHR loops must be OPERABLE. Additionally, one DHR loop must be in operation to provide:

- a. Removal of decay heat;
- b. Mixing of borated coolant to minimize the possibility of criticality; and
- c. Indication of reactor coolant temperature.

An OPERABLE DHR loop consists of a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

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APPLICABILITY

Two DHR loops are required to be OPERABLE, and one in operation in MODE 6, with the water level  $< 23$  ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6, with the water level  $\geq 23$  ft above the top of the reactor vessel flange, are located in LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation—High Water Level."

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ACTIONS

A.1 and A.2

With fewer than the required loops OPERABLE, action shall be immediately initiated and continued until the DHR loop is restored to OPERABLE status or until  $\geq 23$  ft of water level is established above the reactor vessel flange. When the water level is established at  $\geq 23$  ft above the reactor

(continued)

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BASES

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ACTIONS

A.1 and A.2 (continued)

vessel flange, the Applicability will change to that of LCO 3.9.4, and only one DHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions to restore the required forced circulation or water level.

B.1

If no DHR loop is in operation or no DHR loop is OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentration can occur by adding water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

B.2

If no DHR loop is in operation or no DHR loop is OPERABLE, actions shall be initiated immediately and continued without interruption to restore one DHR loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE DHR loops and one operating DHR loop should be accomplished expeditiously.

If no DHR loop is OPERABLE or in operation, alternate actions shall have been initiated immediately under Condition A to establish  $\geq 23$  ft of water above the top of the reactor vessel flange. Furthermore, when the LCO cannot be fulfilled, alternate decay heat removal methods, as specified in the unit's Abnormal and Emergency Operating Procedures, should be implemented. This includes decay heat removal using the charging or safety injection pumps through the Chemical and Volume Control System with consideration for the boron concentration. The method used to remove decay heat should be the most prudent as well as the safest choice, based upon unit conditions. The choice could be different if the reactor vessel head is in place rather than removed.

(continued)

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BASES

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ACTIONS  
(continued)

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one DHR loop is in operation. The flow rate is determined by the flow rate necessary to provide efficient decay heat removal capability and to prevent thermal and boron stratification in the core.

In addition, during operation of the DHR loop with the water level in the vicinity of the reactor vessel nozzles, the DHR loop flow rate determination must also consider the DHR pump suction requirement. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the DHR System in the control room.

SR 3.9.5.2

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

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REFERENCES

1. FSAR, Section [ ].
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## B 3.9 REFUELING OPERATIONS

### B 3.9.6 Refueling Canal Water Level

#### BASES

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##### BACKGROUND

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts, within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident within 10 CFR 100 limits, as provided by the guidance of Reference 3.

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##### APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS and during movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft, and a minimum decay time of 72 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and offsite doses are maintained within allowable limits (Ref. 3).

Refueling canal water level satisfies Criterion 2 of the NRC Policy Statement.

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(continued)

BASES (continued)

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LCO                      A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits as provided by 10 CFR 100.

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APPLICABILITY        LCO 3.9.6 is applicable during CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts, and when moving irradiated fuel assemblies within the containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Fuel Storage Pool Water Level."

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ACTIONS

A.1 and A.2

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or movement of irradiated fuel, action to restore refueling cavity water level must be initiated immediately.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a postulated fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

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REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
  2. FSAR Section [ ].
  3. 10 CFR 100.10.
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11. ABSTRACT (200 words or less)

This report documents the results of the combined effort of the NRC and the industry to produce improved Standard Technical Specifications (STS), Revision 1 for Babcock & Wilcox Plants. The changes reflected in Revision 1 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This NUREG is the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. The improved STS will be used as the basis for individual nuclear power plant licensees to develop improved plant-specific technical specifications. This report contains three volumes. Volume 1 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.9 of the improved STS.

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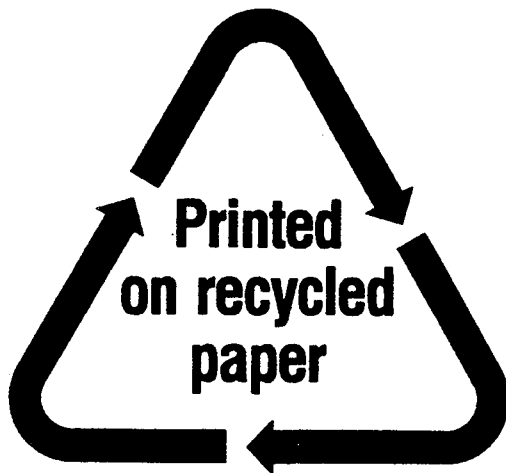
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