NUREG–1431
Vol. 3

Standard Technical Specifications
Westinghouse Plants

Bases (Sections 3.4–3.9)

Draft Report for Comment

Issued by the
U.S. Nuclear Regulatory Commission

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PREFACE

This DRAFT NUREG presents the results of the Nuclear Regulatory Commission (NRC) staff review of the Westinghouse Owners Group (WOG) proposed new Standard Technical Specifications (STS). These new STS were developed based on the criteria in the interim Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated February 6, 1987.

The new STS will be used as bases for developing improved plant-specific technical specifications by individual nuclear power plant owners that have PWRs designed by Westinghouse. The NRC staff is issuing this draft new STS for a 30 working-day comment period. Following the comment period, the NRC staff will analyze comments received, finalize the new STS, and issue them for plant-specific implementation.

Comments should be submitted no later than March 15, 1991, in accordance with the following guidance: The exact wording of each proposed change should be marked in pen and ink on copies of all the affected pages of DRAFT NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Each proposed change should be numbered. Each proposed change should be accompanied with a separate technical justification, cross referenced to the applicable proposed change on the marked up pages.

Submit written comments to: David L. Meyer, Chief, Regulatory Publications Branch, Division of Freedom of Information and Publications Services, Office of Administration, U. S. Nuclear Regulatory Commission, Washington, DC 20555. Hand deliver comments to: 7920 Norfolk Avenue, Bethesda, Maryland, between 7:45 a.m. and 4:15 p.m. on Federal workdays.

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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

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow within the limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady-state envelope of operating conditions. The limits placed on RCS pressure, temperature, and flow ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNBR limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNBR limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNBR limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB-limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB-limited transients analyzed in the plant safety analyses (Ref. 1).

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion of ≥ [1.3]. This is the acceptance limit for the RCS DNB parameters. Changes to the facility which 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 rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, "Control Bank Insertion Limits", LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of [2200] psig and the RCS average temperature limit of [581]°F correspond to analytical limits of [2205] psig and [595]°F used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Interim Policy Statement, because they limit the variations of RCS pressure, temperature, and flow, which are initial conditions in the safety analyses.

LCO

This LCO provides limits on the monitored process variables, pressurizer pressure, RCS average temperature, and RCS total flow rate to ensure that the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB-limited transient.

RCS total flow rate contains a measurement error of [2.0]% based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to [2.1]% for no fouling.

Any fouling that might bias the flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, either

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(continued)

BASES (continued)

LCO (continued)

the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The LCO numerical values for pressure, temperature, and flow are given for the measurement location, but have not been adjusted for instrument error.

APPLICABILITY

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

The limit on pressurizer pressure is not applicable during short-term operational transients such as a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER (RTP) per minute or a THERMAL POWER step increase in excess of 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 less than 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

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

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BASES (continued)

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameters.

RCS total flow rate is not a controllable parameter and is not expected to vary during steady-state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced as required by Required Action B.1 to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2-hour Completion Time for restoration of the parameters is based on providing sufficient time to adjust plant parameters, determine the cause for the off-normal condition, and restore the readings within limits, and is based on plant operating experience.

The Completion Time of Required Action A.1 has been provided with a Note to clarify that all RCS DNB parameters for this LCO are treated as an entity with a single Completion Time, i.e., the Completion Time is on a Condition basis.

RCS pressure, average temperature, and flow rate are considered out of limits if the equipment used to measure these parameters is determined to be inoperable. Required Action A.1 applies to restoring such equipment to OPERABLE status.

B.1

If Required Action A.1 is not met within the Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 2 in 6 hours.

In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The 6 hours specified is a reasonable time [illegible] sily permits the plant power to be reduced at an [illegible] rate in conjunction with even control of steam generator heat removal.

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BASES (continued)

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 of pressurizer 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 12-hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

[For this facility, pressurizer pressure is measured as follows:]

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 of RCS average temperature is sufficient to ensure that the 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 for potential degradation and to verify operation is within safety analysis assumptions.

[For this facility, RCS average temperature is measured as follows:]

SR 3.4.1.3

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

[For this facility, RCS total flow rate is measured as follows:]

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 18 months

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow.

The intent of the Surveillance Frequency of 18 months is to reflect the importance of re-verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

An exception to SR 3.0.4 is claimed for the performance of the precision heat balance required by SR 3.4.1.4. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The surveillance shall be performed within 24 hours after reaching 90% RTP.

REFERENCES

1. [Unit Name] FSAR, Section [15], "[Accident Analysis]."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature For Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.4. In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transients and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil-ductility reference temperature when the reactor is critical.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below hot zero power (HZP) temperature. This band allows critical operation slightly below HZP temperature during plant startup, but this does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP temperature and the minimum temperature for criticality.

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for HZP is a process variable that is an initial condition of DBAs such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed for zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission-product barrier.

The low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 547°F (Ref. 1). These DBA analyses establish the acceptance limits for the minimum temperature for criticality. Reference to the analyses for these DBAs is used to assess changes to the facility which could affect the minimum temperature for criticality as they relate to the acceptance limits.

The RCS minimum temperature for criticality ensures the HZP temperature at criticality is within a small band, is a process variable monitored by the operators during startup, and is displayed on permanently installed instrumentation in the control room. As such, the RCS minimum temperature for criticality satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

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BASES (continued)

APPLICABILITY

In MODES 1 and 2, with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES. In addition, reactor criticality is permitted only when any RCS average loop temperature is > 551°F and the $T_{avg} - T_{ref}$ deviation alarm is in an alarm state. In the range of 541°F, at the minimum temperature for criticality, to 547°F, there is a potential for RCS loop average temperature to fall below the LCO requirement. Below 547°F, T_{ref} is essentially constant and equal to 547°F ($T_{no\ load}$). Therefore, a $T_{avg} - T_{ref}$ deviation alarm would be due to the movement of RCS loop average temperature below $T_{no\ load}$ and would alarm 1°F above the minimum temperature for criticality. As power level increases, each RCS loop average temperature increases to a value far enough above 547°F that the potential for RCS loop average temperatures to fall below 541°F is so diminished that this LCO is no longer applicable.

The special test exception of LCO 3.1.12, "MODE 2 PHYSICS TEST Exceptions," permits PHYSICS TESTS to be performed at ≤ 5.0% of RATED THERMAL POWER with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

ACTIONS

A.1

With RCS loop average temperature below 541°F, restoration is required within 15 minutes. The Completion Time of 15 minutes restricts the period for operation outside the analyzed limits. The Completion Time is reasonable for the operator to accomplish the specified actions.

The RCS minimum temperature for criticality is considered out of limits if the equipment used to measure RCS loop average temperature is determined to be inoperable

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BASES (continued)

ACTIONS (continued)

Required Action A.1 applies to restoring such equipment to OPERABLE status.

A.2

If the Required Action is not met within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in MODE 3 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30-minute period. The basis for the time is derived from the ability to perform the action and the urgency of maintaining the plant within the analyzed range.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified above 541°F within 15 minutes prior to achieving criticality and every 30 minutes thereafter. The 15-minute time period is long enough to allow the operator to adjust temperatures or delay criticality so the LCO will not be violated, thereby providing assurance that the safety analyses are not violated.

When any RCS loop average temperature is less than 547°F and the T_{avg}-T_{ref} deviation alarm is alarming, RCS loop average temperatures could fall below the LCO requirement without additional warning. The SR to verify RCS loop average temperatures every 30 minutes is frequent enough to prevent the inadvertent violation of the LCO.

[For this facility, RCS loop average temperature is measured as follows:]

REFERENCES

1. [Unit Name] FSAR, Section [15.0.3], "[Title]."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

LCO 3.4.3 contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and criticality, and data for the maximum rate of change of reactor coolant temperature.

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

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

10 CFR 50, Appendix G (Ref. 1) requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. An adequate margin to brittle failure must be provided during normal operation, anticipated operational occurrences, and system hydrostatic tests. Reference 1 mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

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BASES (continued)

BACKGROUND (continued)

Reference 1 addresses the concern that undetected flaws can exist in the RCPB components and can result in brittle (non-ductile) failure if subjected to unusual pressure or thermal stresses. Certain RCS P/T combinations can cause stress concentrations at flaw locations, which, in turn, can cause flaw growth and result in failure before the ultimate strength of the material is reached. Material toughness resists and can even arrest flaw growth.

Material toughness varies with temperature and is lower at room temperature than at operating temperature. Toughness also depends on the chemistry and impurities of the base material, weld material, and heat-affected zone material. Furthermore, neutron fluence affects material toughness by decreasing ductility; the effect accumulates, and the portion of the RCPB in a high fluence area, the vessel beltline region, steadily decreases in ductility with exposure time.

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 3). Although any location in the RCPB is subject to non-ductile failure, the more restrictive limits apply to the vessel beltline, the vessel closure head, and the vessel outlet nozzles. With increased neutron fluence, the vessel beltline, with base metals and welds, typically becomes the most restrictive region.

Material toughness properties of the ferritic materials of the reactor vessel are determined in accordance with the NRC Standard Review Plan (Ref. 4), the American Society for Testing Material (ASTM), ASTM E 185 (Ref. 5), and additional reactor vessel requirements. These properties are then evaluated in accordance with Reference 2.

One indicator of the temperature effect on ductility is the nil-ductility temperature (NDT). The NDT is that temperature below which non-ductile fracture failure may occur. Ductile failure may occur above the NDT.

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BASES (continued)

BACKGROUND (continued)

A range of NDT data points for the steel alloy used in reactor vessel fabrication has been established by testing, but the exact value of NDT cannot be determined. Therefore, a nil-ductility reference temperature (RT_{NDT}) has been established by experimental means. The neutron embrittlement effect on the material toughness is reflected by increasing the RT_{NDT} as exposure to neutron fluence increases.

In effect, the temperature below which non-ductile failure can occur increases over time in operation. Reference 3 provides guidance for evaluating the effect of neutron fluence. To assist in evaluating the amount of RT_{NDT} shift to be applied, surveillance specimens, made up of samples of reactor vessel material, are placed near the inside wall of the reactor vessel in the beltline region.

As the RT_{NDT} increases with vessel exposure to neutron fluence and the material toughness decreases, the P/T limit curves are correspondingly adjusted. This gives limits that provide pressure boundary protection over the design life of the vessel. The effect of the RT_{NDT} shift is to cause the pressure limit to decrease at a given temperature.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 5) and Appendix H of 10 CFR 50 (Ref. 6). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

This specification provides two types of limits:

a. RCS P/T curves that define allowable operating regions; and

b. Limits on the allowable rate of change of temperature of the reactor coolant, which affect the thermal gradients through the wall of the vessel and, thus, the tensile stresses in the wall.

In use, the P/T curves are primarily for prevention of non-ductile failure, whereas the limits on rate of change assist in preventing both ductile and non-ductile failures.

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BASES (continued)

BACKGROUND (continued)

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

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

The calculation to generate the ISLH testing curve uses different safety factors (per Ref. 2) than the heatup and cooldown curves. The ISLH testing curve also extends to the RCS design pressure of 2500 psia.

The criticality limit curve includes the Reference 1 requirement that it be no less than 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH Testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature For Criticality."

The 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 non-ductile failure. Although created to provide limits for these specific normal operations, the curves also can be used to determine if an evaluation is necessary for an abnormal transient.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that could have resulted in brittle failure of the RCPB, possibly leading to a non-isolable leak or loss-of-coolant accident. In the event these limits are exceeded, an evaluation must be performed

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BASES (continued)

BACKGROUND (continued)

to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 7) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate-of-change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the RCPB, an unanalyzed condition. Reference 8 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

The analyses comprise a number of steps that establish the limits. Following are the basic elements:

a. Define the temperature profile. The reactor coolant temperature rate of change is defined so that normal plant operation can readily proceed without constraint. Cooldown and ISLH Testing rates of change are similarly defined. These rates of change become LCO limits, as well as the bases for the heat transfer calculations.

b. Perform heat transfer calculations. The results determine the thermal gradient through the vessel wall. The analyses account for variances in flow rate and the consequent changes in the rate of heat transfer between the reactor coolant and the wall during different stages of heatup and cooldown.

c. Establish the material toughness as a function of RT_{NDT}. ASME Code, Section III, Appendix G provides the basis for RT_{NDT}, and Regulatory Guide 1.99 provides the basis for adjusting RT_{NDT} as a function of neutron fluence and material constituents and impurities.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

d. Perform a LEFM analysis to establish the pressure and temperature limits. The criterion for setting the limits is that the combined pressure and temperature stresses cannot exceed the material toughness for the specific temperature under examination. The analytical stress concentration at each location is riven by postulating specific flaw sizes. Stress intensity factors for pressure and temperature are calculated and compared to a reference stress intensity factor. Safety factors are applied to the pressure stress intensity factor.

With the material toughness established as a function of RT_{NDT}, stress analyses are performed per Reference 2 to set the P/T limits. The limiting location of maximum stress may vary during heatup or cooldown operations, depending on pressure, temperature, and temperature rates of change.

Thus, the heatup and cooldown curves are composites of the limiting pressures at specific temperatures, with separate curves derived for varying heatup and cooldown rates.

e. Adjust the curves. The curves are adjusted for differences in elevation between the instrument tap locations and the vessel beltline and for system pressure losses at different stages of heatup or cooldown. The limit curves are also adjusted for the estimated instrument errors of the wide-range pressure and temperature instruments.

The P/T limit curves must account for a requirement from Reference 1 that the minimum temperatures of the closure head flange and vessel flange regions must be at least 120°F above the limiting RT_{NDT} for these regions when the pressure exceeds 20% of the preservice hydrostatic test pressure.

The calculation assumes a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, $\frac{1}{4}$ T, and a length of $\frac{3}{2}$ T exists first at the inside of the vessel wall, then at the outside of the vessel wall. These dimensions are well within the current detection capabilities of inservice inspection techniques. Therefore,

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

the P/T limit curves developed for this postulated defect are conservative and provide adequate protection against non-ductile failure.

To ensure that the radiation embrittlement effects on the RT_{NDT} are accounted for in the calculations for the limit curves, the most limiting RT_{NDT} (of the various reactor vessel components) is used and includes a radiation-induced shift corresponding to the end of the fluence period for which heatup and cooldown curves are generated. This shift is a function of both the neutron fluence and the copper and nickel contents of the vessel material. The heatup and cooldown P/T limit curves include predicted adjustments for the RT_{NDT} shift and state the number of effective full power years for which this shift applies.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operational history by removing and evaluating the irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and at the vessel inside wall are essentially identical, the measured transition shift for a sample can be applied to the adjacent section of the reactor vessel. The limit curves must be recalculated when the actual RT_{NDT} from the surveillance specimens is higher than the calculated RT_{NDT} for the presumed radiation exposure.

RCS P/T limits satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

The two elements of this LCO are:

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

b. Limits on the rate of change of temperature.

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

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BASES (continued)

LCO (continued)

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

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

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of non-ductile (brittle) failure in accordance with 10 CFR 50, Appendix G (Ref. 1). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, they are applicable at all times in keeping with the concern for non-ductile 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 supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow DNB Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for non-ductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

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BASES (continued)

ACTIONS

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures.

A.1 and A.2

Operation outside the P/T limits must be restored to within the limits. The RCPB must be returned to a condition that has been verified by stress analyses. Restoration is in the proper direction to reduce RCPB stress.

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

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with preanalyzed transients in the stress analyses, new analyses, or inspection of the components. The evaluation must be completed, documented, and approved in accordance with established plant procedures and administrative controls.

ASME Code, Section XI, Appendix E (Ref. 6) may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline. The 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 both Required Action A.1 and Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to restore operation within limits and perform the evaluation

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BASES (continued)

ACTIONS (continued)

of the effects of the excursion outside the allowable limits. Restoration alone is insufficient because higher-than-analyzed stresses may have occurred and may have affected the RCPB integrity.

The combination of RCS pressure and temperature is considered out of limits if the equipment used to measure RCS pressure or temperature is determined to be inoperable. Required Action A.1 and Required Action A.2 apply to restoring such equipment to OPERABLE status.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused drastic entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced P/T. In reduced conditions, the possibility of propagation of undetected flaws is decreased.

If the required restoration activity cannot be accomplished in 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce P/T.

If the required evaluation for continued operation cannot be accomplished in 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Requi: ! Action B.1 and Required Action B.2. A favorable evaluation must be completed, documented, and approved before returning to operating P/T conditions.

Pressure and temperature are reduced by placing the plant in MODE 3 within 6 hours and in MODE 5 with RCS pressure < 500 psig in 36 hours.

The 6-hour Completion Time is reasonable by operating experience to reach the required MODE from full power, in an orderly manner and without challenging plant systems.

The 36-hour Completion Time for achieving MODE 5 also considers operating experience to reach the required MODE

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BASES (continued)

ACTIONS (continued)

from full power in an orderly manner and without challenging plant systems. The time permits a soak period, if needed, or a slower cooldown (~5°F/hr). A soak period may be desirable if a temperature rate-of-change limit has been violated.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

Verification that operation is within LCO limits is required every 30 minutes when RCS P/T conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate-of-change limits are specified in hourly increments, 30 minutes permit 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.

A Note requires this Surveillance to be performed only during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2, "RCS Minimum Temperature for Criticality," contains a more restrictive requirement.

[For this facility, RCS P/T are measured as follows:]

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

2. ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure."

3. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

4. NUREG-0800, USNRC Standard Review Plan, Section 5.3.1, "Reactor Vessel Materials," Rev. 1, July 1981.

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BASES (continued)

REFERENCES (continued)

5. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,"July 1982.

6. Title 10, Code of Federal Regulations, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

7. ASME Boiler and Pressure Vessel Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events."

8. WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops—MODES 1 & 2

BASES

BACKGROUND

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

The secondary functions of the RCS include:

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

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

c. Carrying the soluble neutron poison, boric acid;

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

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

The reactor coolant is circulated through [four] loops connected in parallel to the reactor vessel, each containing a SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the cladded fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

Both transient and steady-state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming [four] RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses which are most important to RCP operation are the [four] pump coastdown, single pump locked rotor, single pump [broken shaft or coastdown], and rod withdrawal events.

The above analyses are for Design Basis Accidents (DBAs) that establish the acceptance limits for the RCS loops. Reference to the analyses for these DBAs is used to assess changes to the RCS loops as they relate to the acceptance limits.

Steady-state DNB analysis has been performed for the [four] RCS loop operation. For [four] RCS loop operation, the steady-state DNB analysis, which generates the pressure and temperature safety limit (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 109% RATED THERMAL POWER (RTP). This is the design overpower condition for [four] RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 107% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points which result in a minimum DNBR $\geq$ the critical heat flux correlation limit.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the safety limit, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS loops satisfy Criterion 2 of the NRC Interim Policy Statement, because the RCS flow rate is an initial condition for transient and steady-state analyses.

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BASES (continued)

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, [four] pumps are required at rated power.

An OPERABLE RCS loop is composed of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program. RCS loop OPERABILITY also includes the appropriate instrumentation for flow, level, pressure, and temperature for control, protection, and indication. [For this facility, these specific instrumentation channels are:]

[For this facility, an RCS loop in operation is comprised of the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure RCS loop OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the RCS loops inoperable and their justification are as follows:]

APPLICABILITY

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

Maximum decay heat production is approximately 7% of RTP. As such, the forced circulation flow and heat sink requirements are reduced for lower, non-critical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by LCO 3.4.5 (MODE 3), LCO 3.4.6 (MODE 4), LCO 3.4.7 (MODE 5, Loops Filled), and LCO 3.4.8 (MODE 5, Loops Not Filled).

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BASES (continued)

ACTIONS

A.1

If the required number of RCS loops are not OPERABLE or not in operation, the Required Action is to reduce power and bring the plant to MODE 3. The Required Action 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 in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This Surveillance requires verification that the required number of loops are OPERABLE, in operation and circulating reactor coolant every 12 hours. Verification includes flow rate and temperature monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.4.2

This Surveillance provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program, emphasizes the importance of SG tube integrity. Even though this Surveillance cannot be performed at normal operating conditions, its inclusion in this specification is necessary to invoke the Technical Specification requirement for this important inspection program. The preservice, inservice and, if required, augmented inservice inspections performed at shutdown are to demonstrate SG performance and gauge its reliability. During operating conditions, the best indication of SG operation is the RCS water inventory balance performed in accordance with the requirements of LCO 3.4.13, "RCS Operational LEAKAGE."

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BASES (continued)

REFERENCES 1. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops—MODE 3

BASES

BACKGROUND

In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG) to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through [four] RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the cladded fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

APPLICABLE SAFETY ANALYSES

Whenever the reactor trip breakers are in the closed position, the possibility of an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a control rod drive mechanism housing.

Therefore, in MODE 3 with reactor trip breakers in the closed position, accidental control rod withdrawal from subcritical is postulated and requires at least two RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

the reactor trip breakers are open, two RCS loops are required to be OPERABLE but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

The analysis for the inadvertent rod withdrawal event establishes the acceptance limits for the RCS loops in MODE 3. Reference to the analysis for this event is used to assess changes to the RCS loops as they relate to the acceptance limits.

Failure to provide heat removal may result in challenges to a fission-product barrier. The RCS loops are part of the primary success path which 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. As such, this LCO satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The purpose of this LCO is to require that at least two of the RCS loops be OPERABLE. In MODE 3 with the reactor trip breakers in the closed position, two RCS loops must be in operation. Two RCS loops are required to be in operation in MODE 3 with reactor trip breakers closed due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensure that the Safety Limit criteria will be met for all of the postulated accidents.

With the reactor trip breakers in the open position, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met.

The Note in the LCO permits all RCPs to be de-energized for $\leq$ 1 hour per 8-hour period. The purpose of the Note is to perform tests which are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss-of-flow accident. This test is generally performed in MODE 3 during the initial

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BASES (continued)

LCO (continued)

startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.

The no-flow test may be performed in MODE 3, 4, or 5, and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1-hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be assured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. [For this facility, core outlet temperature is measured as follows:]

The Note is not applicable if the core outlet temperature instrumentation is found inoperable.

An OPERABLE RCS loop is comprised of an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program. RCS loop OPERABILITY also includes the appropriate flow, level, pressure, and temperature instrumentation for control, protection, and indication.

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BASES (continued)

LCO (continued)

[For this facility, these specific instrumentation channels are:] An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

[For this facility, an RCS loop in operation is comprised of the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure RCS loop OPERABILITY in MODE 3:]

[For this facility, those required support systems which, upon their failure, do not require declaring the RCS loops inoperable in MODE 3 and their justification are as follows:]

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with reactor trip breakers in the closed position. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the reactor trip breakers open.

Operation in other MODES is covered by LCO 3.4.4 (MODES 1 and 2), LCO 3.4.6 (MODE 4), LCO 3.4.7 (MODE 5, Loops Filled), and LCO 3.4.8 (MODE 5, Loops Not Filled).

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant non-operating 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 and because of the low probability of a failure in the remaining loop occurring during this period.

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BASES (continued)

ACTIONS (continued)

B.1

If restoration is not possible within 72 hours, the unit must be placed in MODE 4. In MODE 4 the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions without challenging plant systems.

C.1 and C.2

If only one RCS loop is OPERABLE and in operation, and the reactor trip breakers are closed, the Required Action is either to restore an RCS loop to operation or to open the reactor trip breakers. When the reactor trip breakers are in the closed position, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the reactor trip breakers must be opened. The Completion Time of 1 hour for opening the breakers is adequate to perform this operation in an orderly manner without exposing the unit to risk for an undue time period.

D.1, D.2 and D.3

If no RCS loop is OPERABLE or OPERABLE and in operation, except as during conditions permitted by the Note in the LCO section, the reactor trip breakers must be opened, all operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the reactor trip breakers removes the possibility of an inadvertent rod withdrawal. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This Surveillance requires verification that the required number of loops are OPERABLE, in operation, and circulating reactor coolant every 12 hours. Verification includes flow rate and temperature monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to the departure from nucleate boiling. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

SG OPERABILITY is verified by ensuring that the secondary-side wide-range water level is ≥ 17%. If the SG secondary-side wide-range water level is < 17%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12-hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

In the event that the equipment used to measure SG secondary-side wide-range water level is found inoperable, the associated RCS loop(s) are declared inoperable.

[For this facility, SG water level is measured as follows:]

SR 3.4.5.3

Verification that the required number of RCS loops are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that additional RCS 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 availability to the required RCPs. The Note indicates that the Surveillance is not required when reactor trip breakers are closed, because both RCPs are in operation and not in standby. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable through operating experience.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops—MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through [four] RCS loops connected in parallel to the reactor vessel, each loop containing a SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for heat removal (Ref. 1).

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

The analysis for the boron dilution event establishes the acceptance limits for the RCS loops in MODE 4. Reference to the analysis for this event is used to assess changes to the RCS loops as they relate to the acceptance limits.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Failure to provide decay heat removal may result in challenges to a fission-product barrier. Although the RHR System does not meet any specific criterion of the NRC Interim Policy Statement, it was identified in the Policy Statement as an important contributor to risk reduction, and this LCO is thus retained as a Specification.

LCO

The purpose of this LCO is to require that at least two loops, RCS or RHR, be OPERABLE in MODE 4, and that one of these loops be in operation. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to meet the single failure criterion.

Note 1 in the LCO permits all RCPs or RHR pumps to be de-energized for ≤ 1 hour per 8-hour period. The purpose of the Note is to permit tests which are designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no-flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1-hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

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BASES (continued)

LCO (continued)

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. [For this facility, core outlet temperature is measured as follows:]

Note 2 in the LCO requires that the secondary-side water temperature of each SG be ≤ [ ]°F above each of the RCS cold leg temperatures before the start of a RCP with an RCS cold leg temperature ≤ 275°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. [For this facility, SG secondary-side water temperature and RCS cold leg temperature are measured as follows:]

Notes 1 and 2 are not applicable if the core outlet temperature, SG secondary-side water temperature, or RCS cold leg temperature instrumentation is found inoperable.

An OPERABLE RCS loop is comprised of an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.6.2. RCS loop OPERABILITY also includes the appropriate flow, level, pressure, and temperature instrumentation for control, protection, and indication. [For this facility, these specific instrumentation channels are:]

Similarly for the RHR System, an OPERABLE RHR loop is comprised of an OPERABLE RHR pump providing forced flow to an OPERABLE RHR heat exchanger, along with appropriate flow and temperature instrumentation for control, protection, and indication. [For this facility, these specific instrumentation channels are:] RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

[For this facility, an RCS loop in operation is comprised of the following:]

[For this facility, an RHR loop in operation is comprised of the following:]

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BASES (continued)

LCO (continued)

[For this facility, the following support systems are required to be OPERABLE to ensure RCS and RHR loops OPERABILITY in MODE 4:]

[For this facility, those required support systems which, upon their failure, do not require declaring the RCS or RHR loops inoperable in MODE 4 and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of an RHR System and the justification of whether or not each supported system is declared INOPERABLE are as follows:] It should be noted that LCO 3.4.6 may need to be augmented with additional Conditions, if it is determined that the RHR System provides support to other systems included in the Technical Specifications during this MODE of operation.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by LCO 3.4.4 (MODES 1 and 2), LCO 3.4.5 (MODE 3), LCO 3.4.7 (MODE 5, Loops Filled), and LCO 3.4.8 (MODE 5, Loops Not Filled).

ACTIONS

A.1

If only one RCS loop is OPERABLE and in operation and there is no RHR loop OPERABLE, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The Completion Time of 15 minutes emphasizes the importance of maintaining the availability of two paths for heat removal.

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BASES (continued)

ACTIONS (continued)

B.1 and B.2

If only one RHR loop is OPERABLE and in operation and there is no RCS loop OPERABLE, an inoperable RCS or RHR loop must be restored to OPERABLE status to satisfy single failure considerations. The Completion Time of 1 hour is based on the fact that one loop is still available for cooldown for the reduced heat loads of this operating MODE.

If Required Action B.1 cannot be accomplished within the required Completion Time, the unit must be placed in MODE 5 within the following 24 hours. Placing the unit in MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 (≤ 200°F) rather than MODE 4 (200-300°F). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging safety systems.

C.1 and C.2

If no loop is OPERABLE or OPERABLE and in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be suspended, and action to restore one RCS or RHR 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 heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance requires verification that the required number of loops are OPERABLE, in operation, and circulating reactor coolant every 12 hours. Verification includes flow rate and temperature monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to the departure from nucleate boiling. The

(continued)

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

SR 3.4.6.2

SG OPERABILITY is verified by ensuring that the secondary-side wide-range water level is ≥ 17%. If the SG secondary-side wide-range water level is < 17%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12-hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

In the event that the equipment used to measure SG secondary-side wide-range water level is found inoperable, the associated RCS loop(s) are declared inoperable.

[For this facility, SG water level is measured as follows:]

SR 3.4.6.3

Verification that the required number of loops are OPERABLE ensures that additional RCS or RHR 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 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.

REFERENCES

1. Generic Letter 88-17, "Loss of Decay Heat Removal," U.S. Nuclear Regulatory Commission, October 17, 1988.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops—MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is to remove decay heat and transfer this heat to the steam generator(s) (SG(s)) or residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated by means of two RHR loops connected to the reactor vessel, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for both control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR 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 (Ref. 1).

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop which must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary-side water levels above 17% to provide an alternate method for decay heat removal.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

The analysis for the accidental boron dilution event establishes the acceptance limits for the RCS loops in MODE 5. Reference to the analysis for this event is used to assess changes to the RCS loops as they relate to the acceptance limits.

Failure to provide decay heat removal may challenge the integrity of a fission-product barrier. Although the RHR System does not meet any specific criterion of the NRC Interim Policy Statement, it was identified in the Policy Statement as an important contribution to risk reduction, and this LCO was thus retained as a Specification.

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary-side water level $\geq$ 17%. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary-side water levels $\geq$ 17%. Should the operating RHR loop fail, the SGs could be used to remove the decay heat.

Note 1 in the LCO permits all RHR pumps to be de-energized $\leq$ 1 hour per 8-hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no-flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow

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(continued)

BASES (continued)

LCO (continued)

characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1-hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. [For this facility, core outlet temperature is measured as follows:]

Note 2 in the LCO allows one RHR 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 such testing is safe and possible.

Note 3 requires that the secondary-side water temperature of each SG be ≤ [ ]°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ 275°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. [For this facility, SG secondary-side water temperature and RCS cold leg temperature are measured as follows:]

Notes 1 and 2 are not applicable if the core outlet temperature, SG secondary-side water temperature, or RCS cold leg temperature instrumentation is found inoperable.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is OPERABLE and in operation. This Note provides for the transition

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BASES (continued)

LCO (continued)

to MODE 4 where an RCS loop is permitted to be in operation and replaces the heat removal function provided by the RHR loops.

An OPERABLE RHR loop is composed of an OPERABLE RHR pump providing forced flow to an OPERABLE RHR heat exchanger, along with appropriate flow and temperature instrumentation for control, protection, and indication. [For this facility, these specific instrumentation channels are:]

RHR 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.

[For this facility, an RHR loop in operation is comprised of the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure RHR loop OPERABILITY and SG secondary-side water level in MODE 5:]

[For this facility, those required support systems which, upon their failure, do not require declaring the RHR loops or SG secondary-side water inoperable in MODE 5 and their justification are as follows:]

APPLICABILITY

In MODE 5 with loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary-side water level of at least [two] SGs is required to be ≥ [17]%.

Operation in other MODES is covered by LCO 3.4.4 (MODES 1 and 2), LCO 3.4.5 (MODE 3), LCO 3.4.6 (MODE 4), and LCO 3.4.8 (MODE 5, Loops Not Filled).

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

If only one RHR loop is OPERABLE and in operation, and less than two SGs have secondary-side water level < [17]%, redundancy for heat removal is lost. Action must be initiated to restore a second RHR loop to OPERABLE status or to restore the SG levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The Completion Times of 15 minutes emphasizes the importance of maintaining the availability of two paths for heat removal.

SG secondary-side water levels are considered out of limits if the equipment used to verify level is determined to be inoperable and Required Action A.1 or A.2 applies.

B.1 and B.2

If no RHR loop is OPERABLE or OPERABLE and in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be suspended, and action to restore a RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing and to preserve the margin to criticality in this type of operation. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This Surveillance requires verification that the required number of loops are OPERABLE, in operation, and circulating reactor coolant every 12 hours. Verification includes flow rate and temperature monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to the departure from nucleate boiling. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary-side wide-range water levels are ≥ 17% ensures an alternate decay heat removal method in the event that the second RHR loop is not OPERABLE. The Note requires the surveillance when the LCO requirement is being met by use of the SGs. If both RHR loops are OPERABLE, this surveillance is not needed. The 12-hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

In the event that the equipment used to measure SG secondary-side wide-range water level is found inoperable, the associated RCS loop(s) are declared inoperable.

[For this facility, SG water level is measured as follows:]

SR 3.4.7.3

Verification that a second RHR loop is OPERABLE ensures that an additional loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. The Note requires the surveillance when the LCO requirement is being met by the use of RHR loops. If secondary-side water level is ≥ 17% in at least two SGs, this surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. Generic Letter 88-17, "Loss of Decay Heat Removal," U.S. Nuclear Regulatory Commission, October 17, 1988.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops—MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. [Each facility shall define what is meant by "loops not filled" for each SG design. Also, expand this definition and background to cover mid-loop operation concerns expressed in Generic Letter 88-17, "Loss of Decay Heat Removal," (Ref. 1)]. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The 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 RHR pump for decay heat removal and transport. The other intent of this LCO is to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

The analysis for the accidental boron dilution event establishes the acceptance limits for the RCS loops in MODE 5. Reference to the analysis for this event is used to assess changes to the RCS loops as they relate to the acceptance limits.

Failure to provide decay heat removal may result in challenges to a fission-product barrier. Although the RHR System does not meet any specific criterion of the NRC

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	Interim Policy Statement, it was identified in the Policy Statement as an important contributor to risk reduction, and this LCO is thus retained as a specification.

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

Note 1 in the LCO permits all RHR pumps to be de-energized for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and temperature is maintained ≤ [160]°F. The LCO Note prohibits boron dilution or draining operations when RHR forced flow is stopped. [For this facility, RCS temperature is measured as follows:]

Note 1 is not applicable if the RCS temperature instrumentation is found inoperable.

Note 2 in the LCO allows one RHR 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 RHR loop is comprised of an OPERABLE RHR pump providing forced flow to an OPERABLE RHR heat exchanger, along with appropriate flow and temperature instrumentation for control, protection, and indication. [For this facility, these specific instrumentation channels are:]

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

[For this facility, an RHR loop in operation is comprised of the following:]

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BASES (continued)

LCO (continued)

[For this facility, the following support systems are required to be OPERABLE to ensure RHR loop OPERABILITY in MODE 5:]

[For this facility, those required support systems which,upon their failure, do not require declaring the RHR loops inoperable in MODE 5 and their justification are as follows:]

APPLICABILITY

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

Operation in other MODES is covered by LCO 3.4.4 (MODES 1 and 2), LCO 3.4.5 (MODE 3), LCO 3.4.6 (MODE 4), and LCO 3.4.7 (MODE 5, Loops Filled).

ACTIONS

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The Completion Time of 15 minutes emphasizes the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is OPERABLE or OPERABLE and in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be suspended and action to restore a RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for uniform dilution, 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 heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification that at least one loop is OPERABLE, in operation, and circulating reactor coolant every 12 hours. Verification includes flow rate and temperature monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to the departure from nucleate boiling. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.8.2

Verification that the required number of loops are OPERABLE 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 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.

REFERENCES

1. Generic Letter 88-17, "Loss of Decay Heat Removal," U.S. Nuclear Regulatory Commission, October 17, 1988.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

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

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

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. The steam bubble limits the volume of non-condensible gases. Relatively small amounts of non-condensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters assures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and assuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high-pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the

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BASES (continued)

BACKGROUND (continued)

system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single-phase natural circulation and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed in lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of non-condensible gases normally present. The steam bubble limits the volume of non-condensible gases.

Safety analyses presented in the FSAR do not take credit for pressurizer heater operation, however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum level limit is of prime interest for the loss of main feedwater (LOMFW) event. Conservative safety analyses assumptions for this event indicate that it produces the largest increase of pressurizer level caused by a moderate frequency event. Thus, this event has been selected to establish the pressurizer water level limit. Assuming proper response action by emergency systems, the level limit prevents water relief through the pressurizer safety valves. Since prevention of water relief is a goal for abnormal transient operation rather than a safety limit, 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.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The above analyses are Design Basis Analyses (DBAs) which are used to establish acceptance limits for the pressurizer.These DBAs are referenced to assess changes to the pressurizer to evaluate their effect on the acceptance limits.

LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," includes a Pressurizer Water Level—High trip, function 10, which is set at 92% of span, which is also the limit for this LCO. This trip setpoint becomes active when the power level reaches the P-7 setpoint. The Pressurizer Water Level—High trip function provides a backup signal for the Pressurizer Pressure—High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass st[illegible] order to achieve their sign energy removal rate. A trip is actuated prior to the pressurizer becoming [illegible] solid.

The requirement for emergency power supplies is based on NUREG-0737 (Ref. 2). 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 Interim Policy Statement because it prevents exceeding the initial reactor coolant mass which is an input assumption of the LOCA analysis. The maximum water level also permits the pressurizer safety valves to relieve steam for anticipated pressure increase transients, preserving their function for mitigation. Thus, Criterion 3 is also indirectly applicable.

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. 2), is the reason for providing an LCO.

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BASES (continued)

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1656 cubic feet, which is equivalent to 92% of span, ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady-state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ [150] kW, supplied from either the offsite power source or the emergency power source (when offsite power is not available). The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of [150 kW is derived from the use of 12 heaters rated at 12.5 kW each] The needed amount to maintain pressure is dependent on the losses. The required heaters and their controls must be connected to the emergency buses in a manner that will provide redundant power supply capability.

[For this facility, an OPERABLE pressurizer consists of the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure pressurizer OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the pressurizer inoperable and their justification are as follows:]

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid

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BASES (continued)

APPLICABILITY (continued)

pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, the need to maintain the availability of pressurizer heaters and their emergency power supplies is most pertinent. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODES 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1 and A.2

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady-state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level—High trip.

With pressurizer water level above the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. This is done by placing the unit in MODE 3 with the reactor trip breakers open within 6 hours, and placing the unit in MODE 4 within an additional 6 hours. This takes the unit out of the applicable MODES and restores the unit to operation within the bounds of the safety analyses.

Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems and operators. Further pressure and temperature reduction to MODE 4 places the plant in a MODE where the LCO is not applicable. The 12-hour time to reach the non-applicable MODE is reasonable, based on operating experience, for that evolution.

Pressurizer water level is considered out of limits if the equipment used to verify level is determined to be inoperable, and Required Action A.1 applies.

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BASES (continued)

ACTIONS (continued)

B.1

If one group of pressurizer heaters is 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 would be unlikely in this period. Pressure control may be maintained during this time using normal station-powered heaters.

C.1 and C.2

The plant must be placed in a MODE in which the LCO does not apply if one group of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1. The plant is placed in MODE 3 within 6 hours and in MODE 4 within the following 6 hours. The Completion Time of 6 hours to reach MODE 3 is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is based on operating experience and is a reasonable time for an orderly power reduction from full power without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This Surveillance ensures that during steady-state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess 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 heaters are required to ensure pressure control and the capability of natural circulation in case of loss of offsite power. The 150-kW capacity is based on the heat input to compensate for steady state heat losses from the pressurizer

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

and the heat loss caused by the mini-spray flow used to keep the pressurizer spray nozzle cool. The capacity of the heaters is verified by energizing the heaters and measuring circuit current. The Frequency of 92 days is considered adequate to detect heater degradation and has been found to be acceptable through operating experience.

SR 3.4.9.3

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

This Surveillance demonstrates that the heaters can be manually transferred from the normal to the emergency power supply and energized. The Frequency of 18 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power which are based on Reference 2.

REFERENCES

1. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

2. Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants," USNRC, February 1977.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop-type, spring-loaded, self-actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), [2735 psig], which is 110% of the design pressure.

Because the safety valves are totally enclosed and self-actuating, they are considered independent components. The relief capacity for each valve, 380,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures < 275°F, and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

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

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated

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BASES (continued)

BACKGROUND (continued)

accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased LEAKAGE, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load;

d. Loss of normal feedwater; and

e. Loss of all AC power to station auxiliaries.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events 3, 4, and 5 (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

The above analyses are Design Basis Accidents (DBAs) that establish the acceptance limits for the pressurizer safety valves. Reference to the analyses for these DBAs is used to assess changes to the safety valves as they relate to the acceptance limits.

Pressurizer safety valves satisfies Criterion 3 of the NRC Interim Policy Statement.

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BASES (continued)

LCO

The three pressurizer safety valves are set to open at the RCS design pressure (2500 psia), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1%-tolerance requirements (Ref. 2) for lifting pressures above 1000 psig. The limit protected by this specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the safety limit 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.

The Note suspending LCO 3.0.4 and SR 3.0.4 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 good assurance that the valves are OPERABLE near their design conditions. Only one valve at a time will be removed from service for testing. The [54]-hour exception is based on 18-hours outage time for each of the three valves. The 18-hour period is derived from operating experience that hot testing can be performed in this time frame.

[For this facility, an OPERABLE pressurizer safety valve consists of the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure pressurizer safety valve OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the pressurizer safety valves inoperable and their justification are as follows:]

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of three valves is required because the combined capacity is required to keep reactor

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BASES (continued)

APPLICABILITY (continued)

coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 below 275°F and in MODE 5, because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place in 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time, or if more than one pressurizer safety valve is inoperable, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 4 below 275°F within 12 hours. The 6 hours allowed to reach MODE 3 and the 12 hours allowed to reach MODE 4 are reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems. Below 275°F, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested one at a time and

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequency necessary to satisfy the SRs. No additional requirements are specified.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, NB 7614.3.
2. [Unit Name] FSAR, Section [15], "[Title]."
3. WCAP-7769, Rev. 1, "Topical Report on Overpressure Protection for Westinghouse Pressurized Water Reactors," June 1972.
4. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power-Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air-operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and to close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck-open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck-open PORV is, in effect, a small-break loss-of-coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs 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 PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the

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BASES (continued)

BACKGROUND (continued)

PORVs is based on maintaining pressure below the Pressurizer Pressure—High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure (LTOP) Protection System."

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are used in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical. By assuming PORV manual actuation, the primary pressure remains below the high pressurizer pressure trip setpoint, thus the DNBR calculation is more conservative. Events that assume this condition include a turbine trip and the loss of normal feedwater (Ref. 2).

The above analyses are for Design Basis Accidents (DBAs) that establish the acceptance limits for the pressurizer PORVs. Reference to the analyses for these DBAs is used to assess changes to the PORVs as they relate to the acceptance limits.

The PORVs satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO ensures that the PORVs and their associated block valves are OPERABLE for manual operation to mitigate the effects associated with a SGTR.

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BASES (continued)

LCO (continued)

By maintaining two PORVs and their associated block valves OPERABLE, the single-failure criterion is satisfied. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive leakage. Satisfying the LCO helps minimize challenges to fission-product barriers.

[For this facility, an OPERABLE pressurizer PORV and block valve consist of the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure pressurizer PORV and block valve OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the pressurizer PORVs or block valves inoperable and their justification are as follows:]

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. The most likely cause for a PORV LOCA is a pressure-increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 to minimize challenges to the pressurizer safety valves.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in 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, "Low Temperature Overpressure Protection (LTOP) System," addresses the PORV requirements in these MODES.

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BASES (continued)

APPLICABILITY (continued)

The exception for LCO 3.0.4 permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status. Testing is not performed in lower MODES.

ACTIONS

A.1 and A.2

With the PORV(s) inoperable and capable of being manually cycled, either the PORV(s) must be restored or the flow path isolated within 1 hour. The block valve(s) should be closed but power must be maintained to the block valve(s), since removal of power would render the block valve inoperable and Condition C would apply. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for a small-break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Times of 1 hour are based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

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

If one PORV is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the block valve. The Completion Time of 1 hour is reasonable, based on challenges to the PORVs during this time period, and provides the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be placed in a MODE in which the LCO does not apply, as required by Condition D.

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BASES (continued)

ACTIONS (continued)

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

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. 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 place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck-open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable based on the small potential for challenges to the system during this time period and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted an additional Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs are not capable of mitigating an overpressure event when placed in manual control. If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the PORV restored to OPERABLE status. If it cannot be restored within this additional time, the plant must be placed in a MODE in which the LCO does not apply, as required by Condition D.

D.1 and D.2

If the Required Actions of Condition A, B, or C are not met, then the plant must be placed in a MODE in which the LCO does not apply. The plant is placed in at least MODE 3 within 6 hours and in MODE 4 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 6 hours to reach MODE 4 is reasonable, considering that a plant can cool down within that time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

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BASES (continued)

ACTIONS (continued)

E.1, E.2.1, E.2.2, E.2.3, and E.2.4

If more than one PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable based on the small potential for challenges to the system during this time, and provides the operator time to correct the situation. If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having more than one PORV inoperable. If no PORVs are restored within the Completion Time, then the plant must be placed in a MODE in which the LCO does not apply. The plant is placed in at least MODE 3 within 6 hours and in MODE 4 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 6 hours to reach MODE 4 is reasonable, considering that a plant can cool down within that time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

F.1, F.2.1, F.2.2, and F.2.3

If more than one block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control and restore at least one block valve within 2 hours and the remaining block valve within 73 hours. The Completion Times are reasonable based on the small potential for challenges to the system during this time and provide the operator time to correct the situation. If the block valves are restored such that only one block valve is inoperable, then the plant will be in Condition C with the time clock started at the original declaration of having more than one block valve inoperable.

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BASES (continued)

ACTIONS (continued)

G.1 and G.2

If the Required Actions of Condition E or F are not met, then the plant must be placed in a MODE in which the LCO does not apply. The plant is placed in at least MODE 3 within 6 hours and in MODE 4 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 6 hours to reach MODE 4 is reasonable considering that a plant can cool down within that time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the Frequency of 92 days is the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 3). If the block valve is closed to isolate a PORV which is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Surveillance Frequency (92 days). Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status, i.e., completion of the Required Actions fulfills the SR.

SR 3.4.11.2

SR 3.4.11.2 is the performance of a CHANNEL CALIBRATION every 18 months. The CHANNEL CALIBRATION assures that the PORV setpoint is appropriately maintained below the RCS high

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

pressure trip setpoint, and thus, remote from transient pressure challenges. The calibration also assures that the PORV setpoint is below the pressurizer safety valve setpoint, thus limiting challenges to the safety valves. The calibration can only be performed during shutdown. The Frequency of 18 months is based on a typical refueling cycle and industry-accepted practice.

SR 3.4.11.3

Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of a SGTR. The Frequency of 18 months is based on a typical refueling cycle and industry-accepted practice.

SR 3.4.11.4

Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of 18 months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.

SR 3.4.11.5

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

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

REFERENCES

1. Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1977.

2. [Unit Name] FSAR, Section [15.2], "[Title]."

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BASES (continued)

REFERENCES (continued)

3. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating 10 CFR 50, Appendix G (Ref. 1), Pressure and Temperature (P/T) limits. The reactor vessel is the limiting RCPB component for demonstrating such protection. Figure 3.4.12-1 provides the maximum allowable actuation logic setpoints for the power-operated relief valves (PORVs). LCO 3.4.12-1 provides maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only during shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS P/T during heatup and cooldown to not exceed the P/T limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires deactivating all but [one] [high pressure injection (HPI)] pump [and one charging pump] and isolating the accumulators. The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and a vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing-pressure event.

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BASES (continued)

BACKGROUND (continued)

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve and, if needed, until the [HPI] pump is actuated by SI.

The LTOP System for pressure relief consists of two PORVs with reduced lift settings[, or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve,] or a depressurized RCS and a RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to keep from overpressurization for the required coolant input capability.

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The LTOP actuation logic monitors both RCS temperature and RCS pressure and determines when a condition not acceptable in the PTLR limits is approached. The wide-range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide-range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

Figure 3.4.12-1 presents the PORV setpoints for LTOP. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits of Figure 3.4.12-1 ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the system

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BASES (continued)

BACKGROUND (continued)

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.

[RHR Suction Relief Valve Requirements]

[During LTOP MODES, the RHR System is operated for decay heat removal and low pressure letdown control. Therefore, the RHR suction isolation valves are open in the piping from the RCS hot legs to the inlets of the RHR pumps. While these valves are open and the RHR suction valves are open, the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.]

[The RHR suction isolation valves and the RHR suction valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. Auto-closure interlocks are not permitted to cause the RHR suction isolation valves to close. The RHR suction relief valves are spring-loaded, bellows-type water relief valves with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.]

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

An RCS vent to meet the flow capacity requirement requires removing a pressurizer safety valve, removing a PORV's internals 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, to not drain the RCS when open.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding [275]°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about [275]°F and below, overpressure prevention falls to two OPERABLE RCS relief valves 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 the reactor vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

Reference 4 contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

<u>Mass Input Type Transients</u>

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

<u>Heat Input Type Transients</u>

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump startup with temperature asymmetry within the RCS or between the RCS and steam generators.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

1. Ensuring only [one] [HPI] pump [and one charging pump] OPERABLE;

2. Immobilizing the accumulator discharge isolation valves in their closed positions; and

3. Disallowing start of a RCP if secondary temperature is more than [50]°F above primary temperature in any one loop. LCOs 3.4.6, "RCS Loops—MODE 4," and 3.4.7, "RCS Loops—MODE 5, Loops Filled," provide this protection.

The Reference 4 analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only one [HPI] pump [and one charging pump are] is [are] actuated by SI. Thus, the LCO requires only [one] [HPI] pump [and one charging pump] OPERABLE during the LTOP MODES. Since neither one RCS relief valve nor the RCS vent can handle a full SI actuation, the LCO also requires the accumulators isolated.

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of accumulator discharge is over a narrower RCS temperature range ([175]°F and below) than that of the LCO ([275]°F and below). Fracture mechanics analyses established the temperature of LTOP Applicability at [275]°F.

The consequences of a small-break loss-of-coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6) requirements by having [one] [HPI] pump [and one charging pump] OPERABLE and SI actuation enabled.

<u>PORV Performance</u>

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in Figure 3.4.12-1. The setpoints are derived

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of SI actuation of [one] [HPI] pump [and one charging pump]. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

The PORV setpoints from Figure 3.4.12-1 will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst-case single active failure.

[<u>RHR Suction Relief Valve Performance</u>]

[The RHR suction relief valves do not have variable P/T lift setpoints like the PORVs. Analyses must show that one RHR suction relief valve with a setpoint at or between [436.5] and [463.5] psig will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, an RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation no greater than 10% of the rated lift setpoint.]

[Although each RHR suction relief valve may itself meet single failure criteria, its inclusion and location within the RHR System does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for LTOP.]

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

[The RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst-case single active failure.]

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 2.07 square inches is capable of mitigating a limiting allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, SI actuation with [one] HPI pump [and one charging pump] OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

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

The RCS vent is passive and is not subject to active failure.

LTOP System satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

This LCO is required to ensure that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low-temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires [one] [HPI] pump [and one charging pump] OPERABLE and all accumulator discharge isolation valves closed and immobilized. LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," defines SI actuation OPERABILITY for the LTOP MODE 4 small-break LOCA.

[For this facility, an OPERABLE HPI pump and charging pump consists of the following:]

[For this facility, an immobilized accumulator discharge isolation valve consists of the following:]

(continued)

(continued)

BASES (continued)

LCO (continued)

[For this facility, an OPERABLE LTOP actuation instrumentation of system consists of the following:]

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two RCS relief valves, as follows:

   1. Two OPERABLE PORVs; or

   A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by Figure 3.4.12-1 and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

   [2. Two OPERABLE RHR suction relief valves; or]

   [An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valve and its RHR suction valve are open, its setpoint is at or between [436.5] and [463.5] psig, and testing proves its ability to open at this setpoint.]

   [3. One OPERABLE PORV and one OPERABLE RHR suction relief valve; or]

b. A depressurized RCS and a RCS vent.

   A RCS vent is OPERABLE when open with an area of at least [2.07] square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

[For this facility, the following support systems are required to be OPERABLE to ensure LTOP System OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the LTOP System inoperable and their justification are as follows:]

(continued)

BASES (continued)

APPLICABILITY

This LCO is applicable in MODE 4 when the temperature of any RCS cold leg is at or below [275]°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above [275]°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above [275]°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

The Applicability statement is modified by a Note that states accumulator isolation is only required when the accumulator pressure is more than or at the RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve surveillance performed only under these P/T conditions.

ACTIONS

A.1 and [B.1]

With more than [one] [HPI] pump OPERABLE [or more than one charging pump OPERABLE], RCS overpressurization is possible.

To immediately initiate actions to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

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

An accumulator unisolated requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

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ACTIONS (continued)

If isolation is needed and cannot be accomplished in 1 hour, Required Actions D.1 and D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to more than [175]°F, an accumulator pressure of [600] psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit also gives this protection.

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

E.1

In Mode 4 when any RCS cold leg temperature is at or below [275]°F, with one RCS relief valve inoperable, two RCS relief valves must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS relief valves [in any combination of the PORVS and the RHR suction relief valves] are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the RCS relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

F.1

The consequences of operational events that will overpressure the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.

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BASES (continued)

ACTIONS (continued)

G.1

The RCS must be depressurized and a vent must be established within 8 hours when (1) both RCS relief valves are inoperable, or (2) a Required Action and associated Completion Time of Condition A through Condition F is not met, or (3) the LTOP System is inoperable for any reason other than Condition A through Condition F. The vent must be sized at least 2.07 square inches to ensure that the flow capacity is greater than that required for the worst-case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1[, SR 3.4.12.2,] and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, all but [one] [HPI] pump [and all but one charging pump] are verified locked out with power removed and the accumulator discharge isolation valves are verified closed and locked out.

The Frequency of 15 minutes before reducing RCS temperature to less than the LTOP arming temperature ensures that the maximum number of pumps permitted OPERABLE is not exceeded. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to verify the required status of the equipment.

[SR 3.4.12.4]

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.7

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

for the RHR suction isolation valve Surveillance.) This Surveillance is only performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction valve is verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls, such as valve status indications available to the operator in the control room, that ensure that the RHR suction valve remains open.

The ASME Code, Section XI (Ref. 8) test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.5

The RCS vent of at least 2.07 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that cannot be locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is performed if the vent is being used to satisfy the pressure relief requirements of the LCO. [For this facility, the basis for the frequencies is as follows:]

SR 3.4.12.6

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled motor-operated valve. The power to the valve operator is not required removed and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an over-pressure situation.

The 72-hour Frequency is considered adequate in view of other administrative controls, such as valve position indication available to the operator in the control room, that ensure that the PORV block valve remains open.

[SR 3.4.12.7]

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.4 for the RHR suction valve surveillance and for a description of the requirements of the Inservice Testing Program.) This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.

Every 31 days the RHR suction isolation valve is verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The "locked open" valve must be locally verified in its open position with the manual actuator locked in its inactive position. The Frequency is considered adequate in view of the procedural controls governing valve operation and also as a means of providing added assurance of correct valve position.

SR 3.4.12.8

Performance of an ANALOG CHANNEL OPERATIONAL TEST (ACOT) is required within [12] hours after decreasing RCS temperature to ≤ [275]°F and every 31 days on eacn required PORV to verify and, as necessary, adjust its lift setpoint. The ACOT will verify the setpoint is within the allowed maximum of Figure 3.4.12-1. PORV actuation could depressurize the RCS and is not required.

The [12]-hour Frequency considers the unlikelihood of a low temperature overpressure event during the time. [For this facility, the basis for the 31-day Frequency is as follows:]

[For this facility an ACOT consists of the following:]

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

[For this facility, the required number of RCS temperature and RCS pressure channels consist of the following:]

In the event that the instrumentation used to measure RCS temperature and RCS pressure is found inoperable, the associated PORVs are declared inoperable.

A Note makes SR 3.0.4 not applicable. The ACOT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within [12] hours after entering the LTOP MODES.

SR 3.4.12.9

Performance of a CHANNEL CALIBRATION on each PORV actuation channel is required every [18] months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

[For this facility, a CHANNEL CALIBRATION on a PORV actuation channel consists of the following:]

[For this facility, the basis for the Frequency is as follows:]

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operation."

3. ASME Boiler and Pressure Vessel Code, Section III.

4. [Unit Name] FSAR, Section [15.], "[Title]."

5. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

6. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

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BASES (continued)

REFERENCES (continued)

7. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light Water Reactors'."

8. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS. The RCS components, including the portions of the connecting systems out to and including the isolation valves, define the reactor coolant pressure boundary (RCPB).

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 and the methods used to identify and quantify them.

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

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is needed 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 RCPB from degradation and the core from inadequate cooling, in addition to

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BASES (continued)

BACKGROUND (continued)	preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss-of-coolant accident.
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 leak 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 or the staff approved licensing basis (i.e., a small fraction of these limits). The above analyses are for Design Basis Accidents (DBAs) that establish the acceptance limits for the RCS operational LEAKAGE. Reference to the analyses for these DBAs is used to assess changes to the facility which could affect LEAKAGE as they relate to the acceptance limits. RCS operational LEAKAGE satisfies Criterion 2 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

LCO

a. No Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, LEAKAGE being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB.

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 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 through all steam generators of 1 gpm 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.

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BASES (continued)

LCO (continued)

e. Primary-to-Secondary LEAKAGE through One SG

The 500 gallons per day SG limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

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 leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary-to-secondary LEAKAGE in excess of the LCO limits must be reduced 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

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BASES (continued)

ACTIONS (continued)

brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be placed in MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors which tend to degrade the pressure boundary.

The Completion Times of 6 hours to reach MODE 3 and 36 hours to reach MODE 5 from full power are reasonable, based on operating experience, to reach the required conditions from full power in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE within the LCO limits assures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of a RCS water inventory balance. 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, the requirement of SR 3.0.4 is not applicable for performing an RCS inventory balance before entering MODE 4 or MODE 3.

Steady-state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and the surveillance is not required unless steady state is established. For RCS Operational LEAKAGE determination by inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 leak detection in the prevention of accidents.

SR 3.4.13.2

The leaktight integrity of the RCPB is verified by visual inspection. The Inservice Testing Program and operational hydrostatic tests at normal operating pressure are acceptable means of verifying no RCPB LEAKAGE. The [18-month] Frequency is based on the refueling cycle and adequately verifies RCPB integrity.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 30, "Quality Of Reactor Coolant Pressure Boundary."

2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary LEAKAGE Detection Systems," May 1973.

3. [Unit Name] FSAR, Section [15], "[Accident Analysis]."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50 Appendix A (Refs. 1, 2, and 3) define RCS PIVs as any two normally closed valves in series within the RCS pressure boundary which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration.

The RCS PIV 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 an inventory balance (SR 3.4.13.1) and identification of excessive unidentified LEAKAGE while implementing Required Action A.1 of LCO 3.4.13. 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, which could degrade the ability for low pressure injection.

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

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BASES (continued)

BACKGROUND (continued)

A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs. This later study concluded that periodic leakage testing of the PIVs can substantially reduce intersystem LOCA probability.

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

a. Residual Heat Removal (RHR) System;

b. Safety Injection System; and

c. Chemical and Volume Control System.

The PIVs are listed in Reference 6.

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

APPLICABLE SAFETY ANALYSES

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 RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCS pressure boundary, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for [600] psig, overpressurization failure of the RHR 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.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Leakage from the PIVs is a factor in the dose rates that are used in safety and accident analyses. Therefore, the leakage must be maintained within LCO limits to ensure assumptions used in the analyses are valid.

These analyses establish the acceptance limits for RCS PIV leakage. Reference to these analyses is used to assess changes to the facility which could affect RCS PIV leakage, as they relate to the acceptance limits.

Since it must be included as part of identified LEAKAGE covered by LCO 3.4.13, "RCS Operational LEAKAGE," RCS PIV leakage satisfies Criterion 2 of the NRC Interim Policy Statement.

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.

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

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

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BASES (continued)

LCO (continued)

diminish the overall leakage channel opening. In such cases, the observed rate is adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one-half power.

[For this facility, the supported systems impacted by the inoperability of an RCS PIV flow path (i.e., it is not within leakage limits or is isolated) and the justification for whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized.

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.

A Note is added to provide clarification that each flow path is treated as an independent entity with an independent Completion Time.

ACTIONS

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

Four hours are provided to reduce leakage in excess of the allowable limit. The period permits operation to continue under stable conditions while leakage is assessed and corrective actions are taken. The 4 hours allow the actions and restrict the operation with leaking isolation valves.

Alternatively, the flow path must be isolated by two valves. Required Actions A.2.1 and A.2.2 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.

Required Action A.2.1 requires that the initial isolation with one valve must be performed within 4 hours. This 4-hour Completion Time is based on the same rationale as the time for Required Action A.1.

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(continued)

BASES (continued)

ACTIONS (continued)

Required Action A.2.2 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.

If the affected flow path is isolated in accordance with Required Actions A.2.1 and A.2.2, the affected flow path must be verified to be isolated on a periodic basis. This is necessary to ensure that the integrity of the fission-product barriers is maintained. The Completion Time of once per 31 days for verifying that each affected flow path is isolated is appropriate because the valves are operated under administrative control and the probability of their misalignment is low.

RCS PIV leakage is considered out of limits if the equipment used to measure RCS PIV leakage is determined to be inoperable at the time SR 3.4.14.1 is performed. Required Action A.1 or Required Actions A.2.1 and A.2.2 apply to restoring such equipment to OPERABLE status.

B.1

With one or more RCS PIVs inoperable in one or more flow paths (i.e., they are not within leakage limits or are isolated), verify that the Required Actions have been initiated for those supported systems declared inoperable by the support RCS PIV flow path within a Completion Time of [ ] hours.

The [ ]-hour Completion Time is defined as the most limiting of all Required Actions for all the supported systems that needed to be declared inoperable upon the failure of one or more support features specified under Condition B.

Required Action B.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of RCS PIV flow paths have been initiated. The can be accomplished by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Action for Condition B of this LCO.]

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

[For this facility, the identified supported systems Required Actions are as follows:]

C.1

With one or more RCS PIVs inoperable in one or more flow paths, and one or more RCS PIVs inoperable in the redundant flow paths, or one or more required support features inoperable associated with the redundant flow paths, this results in the loss of functional capability and LCO 3.0.3 must be immediately entered. However, if the support or supported feature LCO, or both, takes into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered. [For this facility, support systems associated with each RCS PIV flow path supported system are as follows:]

D.1 and D.2

If leakage cannot be reduced, the system isolated, or other Required Actions accomplished within the respective Completion Times, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in MODE 3 within 6 hours and MODE 5 within 36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The Completion Times are reasonable based on operating experience to achieve the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.2.1 or A.2.2 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Testing is to be performed every 9 months, but may be extended up to a maximum of [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 required in 10 CFR 50.55a(g) (Ref. 8), is within the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 9), and is based on the prudence of performing surveillances like this only during an outage. The Surveillance needs stable conditions and has the potential for an unplanned plant transient if performed with the plant at power.

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

SR 3.0.4 is exempted for entry into MODES 3 and 4 to permit leakage testing at high differential pressures with stable conditions not possible in the lower MODES.

[For this facility, RCS PIV or other high pressure isolation valve leakage is measured as follows:]

<u>SR 3.4.14.2 and SR 3.4.14.3</u>

Verifying that the RHR auto-closure interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR 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 less than [425] psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The 18-month Frequency was developed considering the plant conditions needed to perform the surveillance. The 18-month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

(continued)

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

[For this facility, the required number of instrumentation channels associated with the RHR System auto-closure interlock consist of the following:]

In the event that the required number of RHR System auto-closure interlock channels are found inoperable or the SRs are not met, the associated RCS PIVs are declared inoperable, which is equivalent to stating that the valves are not within acceptable leakage limits.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Section 50.2, "Definitions—Reactor Coolant Pressure Boundary."

2. Title 10, Code of Federal Regulations, Part 50, Section 50.55a, "Codes and Standards," Subsection (c), "Reactor Coolant Pressure Boundary."

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, Section V, "Reactor Containment," General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

4. U.S. Nuclear Regulatory Commission (NRC), "Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Appendix V, WASH-1400 (NUREG-75/014), October 1975.

5. U.S. NRC, "The Probability of Intersystem LOCA: Impact Due to Leakage Testing and Operational Changes," NUREG-0677, May 1980.

6. [Document containing list of PIVs.]

7. ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants," paragraph IWV-3423(e).

8. Title 10, Code of Federal Regulations, Part 50, Section 50.55a, "Codes and Standards," Subsection (g), "Inservice Inspection Requirements."

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BASES (continued)

REFERENCES (continued)

9. ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants," Paragraph IWV-3422.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS LEAKAGE Detection Instrumentation

BASES

BACKGROUND

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

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

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE is [(or) and air cooler condensate are] instrumented to alarm for increases of 0.5 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

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

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BASES (continued)

BACKGROUND (continued)

indicator of potential RCS LEAKAGE. A 1° increase in dew point is well within the sensitivity range of available instruments.

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

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.

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 that the transport delay time of the LEAKAGE from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting, as well as monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

RCS LEAKAGE detection instrumentation satisfies Criterion 1 of the NRC Interim Policy Statement.

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 is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor, in combination with a particulate or gaseous activity monitor [and a containment air cooler condensate flow rate monitor], provides an acceptable minimum.

[For this facility, OPERABLE LEAKAGE detection instrumentation consists of the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure LEAKAGE detection instrumentation OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the LEAKAGE detection instrumentation inoperable and their justification are as follows:]

APPLICABILITY

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

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

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BASES (continued)

ACTIONS

A.1 and A.2

With the containment sump monitor inoperable, no form of grab sample could 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, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect LEAKAGE.

Restoration of the sump monitor to OPERABLE status is required to regain the function in a Completion Time of 30 days after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS inventory balance required by Action A.1.

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

With both gaseous and 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 an 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 ready form of leak detection.

C.1, and C.2

With the containment air cooler condensate flow rate monitor inoperable, alternative action is again required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a sample is obtained and analyzed or an inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the

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(continued)

BASES (continued)

ACTIONS (continued)

Containment air cooler condensate flow rate monitor to OPERABLE status.

The 24-hour interval provides periodic information that is adequate to detect RCS LEAKAGE.

D.1 and D.2

With the containment atmosphere radioactivity monitor and the containment air cooler condensate flow rate monitor inoperable, the only means of detecting leakage is the containment sump monitor. This condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30-day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

E.1 and E.2

If a Required Action of Condition A, B, C, or D cannot be met within the required Completion Time, the reactor must be placed in a MODE in which the LCO does not apply. This requires placing the reactor 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 perform the actions in an orderly manner and without challenging plant systems.

F.1

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

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1, SR 3.4.15.2, and SR 3.4.15.3

These SRs are the performance of a CHANNEL CHECK of each of the RCS LEAKAGE detection monitors. The check gives reasonable confidence that each channel is operating properly. The Frequency of 12 hours is based on instrument

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(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

reliability and is reasonable for detecting off-normal conditions. For this facility, a CHANNEL CHECK consists of [ ].

SR 3.4.15.4, SR 3.4.15.5, and SR 3.4.15.6

These SRs are the performance of an ANALOG CHANNEL OPERATIONAL TEST [a CHANNEL FUNCTIONAL TEST] on each of the RCS LEAKAGE detection monitors. 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 31 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

For this facility, an ANALOG CHANNEL OPERATIONAL TEST [a CHANNEL FUNCTIONAL TEST] consists of [ ].

SR 3.4.15.7, SR 3.4.15.8, and SR 3.4.15.9

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

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, Section IV, "Fluid Systems," General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary."

2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," U.S. Nuclear Regulatory Commission.

3. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). 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 values. 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 a SGTR accident were an appropriately small fraction of the 10 CFR 100 guideline dose limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2-hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guidelines (Ref. 1) following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.7, "Secondary Specific Activity."

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a preaccident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/E μCi/gm for gross specific activity.

The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power-operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of a SGTR accident are within a small fraction of the Reference 1 dose guideline values. 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 for more than 48 hours. The safety analysis has concurrent and preaccident iodine spiking levels up to 60.0 μCi/g DOSE EQUIVALENT I-131.

The remainder of the above-limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

the established 48-hour time limit. The occurrence of a SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline values.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the total specific activity in the primary coolant is limited to the number of μCi/gm equal to 100 divided by E (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2-hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2-hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

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

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and total specific activity are necessary to contain the potential consequences of a SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F

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BASES (continued)

APPLICABILITY (continued)

and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

ACTIONS

A.1 and A.2

With the 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 change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the set points of the main steam safety valves, and prevents venting the SG to the environment in an SGTR event. The Completion Time of 6 hours is reasonable, based on operating experience to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

Gross specific activity is considered out of limits if the equipment used to measure gross specific activity is determined to be inoperable at the time SR 3.4.16.1 is performed. Required Action A.1 and Required Action A.2 apply to restoring such equipment to OPERABLE status.

B.1 and B.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals not to exceed 4 hours must be taken to demonstrate that 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 is to continue to provide a trend.

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

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BASES (continued)

ACTIONS (continued)

DOSE EQUIVALENT I-131 specific activity is considered out of limits if the equipment used to measure DOSE EQUIVALENT I-131 specific activity is determined to be inoperable at the time SR 3.4.16.2 is performed. Required Action B.1 and Required Action B.2 apply to restoring such equipment to OPERABLE status.

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

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

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

SR 3.4.16.2

This Surveillance is performed to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The

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(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

14-day Frequency is adequate to trend changes in the iodine-activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change of greater than or equal to 15% RATED THERMAL POWER 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

A radiochemical analysis for $\bar{E}$ determination is required 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 specified 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.

SR 3.0.4 does not apply, so sampling can be performed in MODE 1. The sample must be taken after 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that 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.

[For this facility, DOSE EQUIVALENT I-131 specific activity is measured as follows:]

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," 1973.

2. [Unit Name] FSAR, Section [15.6.3], "[Title]."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 RCS Loop Isolation Valves

BASES

BACKGROUND

The RCS may be operated with loops isolated in order to perform maintenance. While operating with a loop isolated, there is potential for inadvertently opening the isolation valves in the isolated loop. In this event, the coolant in the isolated loop would suddenly begin to mix with the coolant in the operating loops. This situation has the potential of causing a positive reactivity addition with a corresponding reduction of SHUTDOWN MARGIN (SDM) if:

a. The temperature in the isolated loop is lower than the temperature in the operating loops (cold water incident); or

b. The boron concentration in the isolated loop is lower than the boron concentration in the operating loops (boron dilution incident).

As discussed in the FSAR (Ref. 1), the startup of an isolated loop is performed in a controlled manner that virtually eliminates any sudden positive reactivity addition from cold water and/or boron dilution because:

a. LCO 3.4.18, "RCS Isolated Loop Startup," and plant operating procedures require that the boron concentration in the isolated loop be maintained higher than the boron concentration of the operating loops, thus eliminating the potential for introducing coolant from the isolated loop that could dilute the boron concentration in the operating loops;

b. The cold leg loop isolation valve cannot be opened unless the temperatures of both the hot and cold legs of the isolated loop are within 20°F of the temperatures of the hot and cold legs of the operating loops (compliance is ensured by operating procedures and automatic interlocks); and

c. Other automatic interlocks, all of which are part of the Reactor Protection System, prevent opening the hot leg loop isolation valve unless the cold leg loop isolation valve is fully closed.

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BASES (continued)

APPLICABLE SAFETY ANALYSES	During startup of an isolated loop in accordance with LCO 3.4.18, "RCS Isolated Loop Startup," the cold leg loop isolation valve interlocks and operating procedures prevent opening of the valve until the isolated loop and operating loop boron concentrations and temperatures are equalized. This ensures that any undesirable reactivity effect from the isolated loop does not occur. The safety analyses assume a minimum SDM as an initial condition for Design Basis Accidents (DBAs) (Ref. 1). Violation of the LCO, combined with mixing of the isolated loop coolant into the operating loops, could result in the SDM being less than that assumed in the safety analyses. The above analyses are for DBAs that establish the acceptance limits for the RCS loop isolation valves. Reference to the analyses for these DBAs is used to assess changes to the RCS loop isolation valves as they relate to the acceptance limits. The boron concentration of an isolated loop may affect SDM and therefore RCS loop isolation valves satisfies Criterion 2 of the NRC Interim Policy Statement.
LCO	This LCO ensures that a loop isolation valve that becomes closed in MODES 1 through 4 is fully isolated and the plant placed in MODE 5. Loop isolation valves are used for performing maintenance when the plant is in MODE 5 or 6, and startup of an isolated loop is covered by LCO 3.4.18, "RCS Isolated Loop Startup."
APPLICABILITY	In MODES 1 through 4, this LCO is applicable when unisolating an isolated loop with a boron concentration less than that of the operating loops may cause an inadvertent criticality. In MODES 5 and 6, the SDM of the operating loops is large enough to permit operation with isolated loops. In these MODES, controlled startup of isolated loops is possible without significant risk of inadvertent criticality.

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BASES (continued)

ACTIONS

A.1

If power is inadvertently restored to the loop isolation valve operators, the potential exists for accidental isolation of a loop with a subsequent inadvertent startup of the isolated loop. The loop isolation valves have motor operators. Therefore, these valves will maintain their last position when power is removed from the valve operator. With power applied to the valve operators, only the interlocks prevent the valve from being operated. Although operating procedures and interlocks make the occurrence of this event unlikely, the prudent action is to remove power from the loop isolation valve operators. The Completion Time of 30 minutes to remove power from the isolation valve operators is sufficient considering the complexity of the task.

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

Should a loop isolation valve be closed in MODES 1 through 4, the affected loop must be fully isolated immediately and the plant placed in MODE 5 to preclude inadvertent startup of the loop and the potential inadvertent criticality. The Completion Time of Required Action B.1 allows time for borating the operating loops to a shutdown boration level such that the plant can be placed in MODE 4 within 6 hours and MODE 5 within the next 30 hours. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 4 from full power in an orderly manner and without challenging safety systems or operators. Similarly, the Completion Time of 36 hours to reach MODE 5 is reasonable considering that a plant can easily cool down in such a time frame on one safety system train.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

The Surveillance is performed at least once per 31 days to ensure that the loop isolation valves are open, with power removed to the valve operators. The primary function of this Surveillance is to ensure that power is removed

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	from the valve operators, since SR 3.4.4.1 of LCO 3.4.4, "RCS Loops—MODES 1 and 2," ensures that the loop isolation valves are open by verifying every 12 hours that all loops are operating and circulating reactor coolant. The Frequency of 31 days ensures that the required flow can be made available, is based on engineering judgment, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 31-day Surveillance Frequency is justified.
REFERENCES	1. [Unit Name] FSAR, Section [15.2.6], "[Title]."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 RCS Isolated Loop Startup

BASES

BACKGROUND

The RCS may be operated with loops isolated in MODES 5 and 6 in order to perform maintenance. While operating with a loop isolated, there is potential for inadvertently opening the isolation valves in the isolated loop. In this event, the coolant in the isolated loop would suddenly begin to mix with the coolant in the operating loops. This situation has the potential of causing a positive reactivity addition with a corresponding reduction of SHUTDOWN MARGIN (SDM) if:

a. The temperature in the isolated loop is lower than the temperature in the operating loops (cold water incident); or

b. The boron concentration in the isolated loop is lower than the boron concentration in the operating loops (boron dilution incident).

As discussed in the FSAR (Ref. 1), the startup of an isolated loop is done in a controlled manner that virtually eliminates any sudden reactivity addition from cold water or boron dilution because:

a. This LCO and plant operating procedures require that the boron concentration in the isolated loop be maintained higher than the boron concentration of the operating loops, thus eliminating the potential for introducing coolant from the isolated loop that could dilute the boron concentration in the operating loops.

b. The cold leg loop isolation valve cannot be opened unless the temperatures of both the hot leg and cold leg of the isolated loop are within 20°F of the operating loops. Compliance with the temperature requirement is ensured by operating procedures and automatic interlocks.

c. Other automatic interlocks prevent opening the hot leg loop isolation valve unless the cold leg loop isolation valve is fully closed. All of the interlocks are part of the Reactor Protection System.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

During startup of an isolated loop, the cold leg loop isolation valve interlocks and operating procedures prevent opening the valve until the isolated loop and operating loop boron concentrations and temperatures are equalized. This assures that any undesirable reactivity effect from the isolated loop does not occur.

The safety analyses assume that a minimum SDM as an initial condition for Design Basis Accidents (DBAs). Violation of this LCO could result in the SDM being reduced in the operating loops to less than that assumed in the safety analyses.

The boron concentration of an isolated loop may affect SDM and therefore this LCO satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

Loop isolation valves are used for performing maintenance when the plant is in MODE 5 or 6. This LCO ensures that the loop isolation valves remain closed until the differentials of temperature and boron concentration between the operating loops and the isolated loops are within acceptable limits.

APPLICABILITY

In MODES 5 and 6, the SDM of the operating loops is large enough to permit operation with isolated loops. Controlled startup of isolated loops is possible without significant risk of inadvertent criticality. This LCO is applicable under these conditions.

ACTIONS

A.1 and A.2

Required Actions A.1 and A.2 assume that the prerequisites of the LCO are not met and a loop isolation valve has been inadvertently opened. Therefore, the Actions require immediate closure to preclude a boron dilution event or a cold water event. However, each Required Action is preceded

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BASES (continued)

ACTIONS (continued)

by a Note that states that Action is required only when a specific Condition, concentration, or temperature, is not met.

SURVEILLANCE REQUIREMENTS

SR 3.4.18.1

Surveillance is performed to establish that the temperature differential between the isolated loop and the operating loops is ≤ 20°F. Performing the Surveillance 30 minutes prior to opening the cold leg isolation valve provides reasonable assurance that the temperature differential will stay within limits until the cold leg isolation valve is opened and is based on engineering judgment. This Frequency has been shown to be acceptable through operating experience. Since the temperature differential is plant specific, each plant will verify the actual differential temperature.

SR 3.4.18.2

To ensure that the boron concentration of the isolated loop is ≥ the boron concentration of the operating loops, a Surveillance is performed 2 hours prior to opening either the hot or cold leg isolation valve. Performing the Surveillance 2 hours prior to opening either the hot or cold leg isolation valve provides reasonable assurance that the boron concentration difference will stay within acceptable limits until the loop is unisolated. This Frequency has been shown to be acceptable through operating experience.

REFERENCES

1. [Unit Name] FSAR, Section [15.2.6], "[Title]."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 <u>RCS Loops—Test Exceptions</u>

BASES

BACKGROUND

The primary purpose of this test exception is to provide an exception to LCO 3.4.4, "RCS Loops—Modes 1 and 2," to permit reactor criticality under no flow conditions during certain PHYSICS TESTS (natural circulation) demonstration, station blackout, and loss of offsite power) to be performed while at low THERMAL POWER levels. Section XI of 10 CFR Part 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant as specified in GDC 1, Quality Standards and Records (Ref. 2).

The key objectives of a test program are to provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the facility has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations.

The tests [Each faulty shall specify the tests] will include [verifying the ability to establish and maintain natural circulation following a plant trip between 10% and 20% of RATED THERMAL POWER (RTP), performing natural circulation cooldown on emergency power, and during the cooldown showing that adequate boron mixture occurs and that pressure can be controlled using auxiliary spray and pressurizer heaters powered from the emergency power sources].

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BASES (continued)

APPLICABLE SAFETY ANALYSES	The tests described above require operating the plant without forced convection flow and as such are not bounded by any safety analyses. However, operating experience has demonstrated this exception to be safe under the present applicability. The NRC Interim Policy Statement allows the test exceptions to be included as part of the LCOs which they affect. This LCO was retained as a separate LCO for clarity.
LCO	This LCO provides an exemption to the requirements of LCO 3.4.4, "RCS Loops—MODES 1 and 2." The LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without the LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODES 1 and 2), and the appropriate tests could not be performed. In MODE 2 where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER $\leq$ P-7 and the reactor trip setpoints of the OPERABLE power level channels are set at $\leq$ 25% RTP. This ensures, if some problem caused the plant to enter MODE 1 and start increasing plant power, the Reactor Trip System (RTS) would automatically shut it down before power became too high, and thereby prevent violation of fuel design limits. The exemption is allowed even though there are no bounding safety analyses. However, these tests are performed under close supervision during the test program and provide valuable information on the plant's capability to cool down without offsite power available to the reactor coolant pumps.
APPLICABILITY	This LCO is applicable when performing low power PHYSICS TESTS without any forced convection flow. This testing is performed to establish that heat input from nuclear heat

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BASES (continued)

APPLICABILITY (continued)

does not exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.

ACTIONS

A.1

When THERMAL POWER is ≥ the P-7 interlock setpoint 10%, the only acceptable action is to ensure the reactor trip breakers are opened immediately in accordance with Required Action A.1 to prevent operation of the fuel beyond its design limits. Opening the reactor trip breakers will shutdown the reactor and prevent operation of the fuel outside of its design limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.19.1

Verification that the power level is ≤ the P-7 interlock setpoint, 10% will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Plant operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.

SR 3.4.19.2

The power range and intermediate range neutron detectors and the P-7 interlock setpoint must be verified to be OPERABLE and adjusted to the proper value. An ANALOG CHANNEL OPERATIONAL TEST is performed within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The time limit of 12 hours is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

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BASES (continued)

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plant and Fuel Reprocessing Plants."

2. Title 10 Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," 1988.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

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

The blowdown phase of a large-break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission-product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

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

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor-operated isolation valve and two check valves in series. The motor-operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to ensure that the valves will automatically open as RCS pressure increases to above the permissive circuit P-11 setpoint.

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BASES (continued)

BACKGROUND (continued)

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. The valves will automatically open, however, as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium-water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large- and small-break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The loss of offsite power assumption is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large-break LOCA is a double-ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large-break LOCA.

The worst-case small-break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and centrifugal charging pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 3) will be met following a LOCA:

a. Maximum fuel element cladding temperature is $\leq$ 2200°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 that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

For both the large- and small-break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied once discharged. For small breaks, an increase in water volume is a peak clad temperature penalty. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill out the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of [ ] and [ ]. To allow for instrument accuracy, values of [7853] gallons and [8171] gallons are specified.

The minimum boron concentration setpoint is used in the post-LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post-LOCA environment. Of particular interest is the large-break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post-LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large- and small-break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

The accumulators satisfy Criterion 3 of the NRC Interim Policy Statement.

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BASES (continued)

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spills through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open with power removed, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

[For this facility, the following support systems are required to be OPERABLE to ensure accumulator OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the accumulators inoperable and their justification are as follows:]

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures $\leq$ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F.

In MODE 3, with RCS pressure $\leq$ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor-operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

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BASES (continued)

APPLICABILITY (continued)	A Note has been included to provide clarification that Conditions A and B for this LCO are treated as an entity with a single Completion Time.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large-break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break for the majority of plants. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

The ECCS accumulator boron concentration is considered out of limits if the equipment used to verify concentration is determined to be inoperable at the time SR 3.5.1.4 is performed. Required Action A.1 applies to restore such equipment to OPERABLE status.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1-hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen

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BASES (continued)

ACTIONS (continued)

cover pressure ensures that prompt action is taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

The ECCS accumulator borated water volume and nitrogen cover pressure are considered out of limits if the equipment used to verify these parameters is determined to be inoperable at the time SR 3.5.1.2 (volume) or SR 3.5.1.3 (pressure) is performed. Required Action B.1 applies to restore such equipment to OPERABLE status.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in MODE 3 within 6 hours and by reducing pressurizer pressure to $\leq$ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience related to the time required to reach the required plant conditions from full power in an orderly manner and without challenging plant systems.

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification that each accumulator valve is fully open, as indicated in the control room, ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor-operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. A 12-hour Frequency is considered reasonable in view of other

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

administrative controls, such as valve position indications, available to the operator that ensure that a mispositioned isolation valve will be quickly identified.

SR 3.5.1.2 and SR 3.5.1.3

Verification every 12 hours of each accumulator's borated water volume and nitrogen cover pressure is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12-hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off-normal trends. In addition, alarms also signify off-normal conditions.

[For this facility, an accumulator's borated water volume and nitrogen cover pressure are measured as follows:]

SR 3.5.1.4

Thirty-one days for verification to determine that each accumulator's boron concentration is within the required limits is reasonable because the static design of the accumulators limits the ways in which the concentration can be changed. The 31-day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling within 6 hours after a 1% volume increase will identify whether inleakage 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 refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements.

[For this facility, an accumulator's boron concentration is measured as follows:]

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator ensures that an active failure could not result in the undetected closure of an accumulator motor-operated isolation valve. If this were to

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Installation and removal of the breakers is conducted under administrative control. Since this surveillance is a verification that power is removed, a surveillance that is relatively easy, the 31-day Frequency was chosen to provide additional assurances that the breakers are removed.

This SR is modified by a Note that allows power to be supplied to the motor-operated isolation valves when pressurizer pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit 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 SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

2. [Unit Name] FSAR, Section [6], "[Emergency Core Cooling System]."

3. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Plants."

4. [Unit Name] FSAR, Section [15], "[Accident Analysis]."

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS—Operating

BASES

BACKGROUND

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

a. Loss-of-coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately 24 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each

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BASES (continued)

BACKGROUND (continued)

subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the centrifugal charging pumps, the RHR pumps, heat exchangers, and the SI pumps. Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Separate piping supplies each subsystem and each train within the subsystem. The discharge from the centrifugal charging pumps combines prior to entering the boron injection tank (BIT) (if the plant utilizes a BIT) and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. Control valves are set to balance the flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the centrifugal charging pumps supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the other ECCS pumps. Initially,

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BASES (continued)

BACKGROUND (continued)

recirculation is through the same paths as the injection phase. Subsequently, recirculation alternates injection between the hot and cold legs.

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

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.

The ECCS subsystems are actuated upon receipt of an SI 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 buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

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

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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

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

d. Core is maintained in a coolable geometry; and

e. Adequate long-term core cooling capability is maintained.

The LCO also limits the potential for a post-trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large-break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small-break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large-break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and

b. A small-break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization,

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small-break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either 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 a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically 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 RWST 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.

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BASES (continued)

LCO (continued)

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

[For this facility, the following support systems are required to be OPERABLE to ensure ECCS train OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the ECCS trains inoperable and their justification are as follows:]

APPLICABILITY

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

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS—Shutdown."

As indicated in Note 1, the flow path may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room.

As indicated in Note 2, LCO 3.0.4 and SR 3.0.4 are excepted for entry into MODE 3. This exception is required for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," requires that certain pumps be rendered inoperable at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the

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BASES (continued)

APPLICABILITY (continued)

inoperable pumps to OPERABLE status. This Note provides the needed time to restore the pumps and ensures that they will be restored in a timely manner by imposing a time and temperature limit on the actions.

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 3.9.5, "RHR and Coolant Circulation—High Water Level," and LCO 3.9.6, "RHR and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With one or more components inoperable and at least 100% of the SI 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 an NRC reliability evaluation (Ref. 6) and is a reasonable time for repair of many ECCS components.

An ECCS flow path is inoperable if it is not capable of delivering design flow to the RCS. The individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

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 SI flow equivalent to a single OPERABLE ECCS train remains available. This allows increased flexibility in unit operations under circumstances when components in opposite trains are inoperable.

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BASES (continued)

ACTIONS (continued)

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. 6) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

The Completion Time of Required Action A.1 has been provided with a Note to clarify that all ECCS components for this LCO are treated as an entity with a single Completion Time, i.e., the Completion Time is on a Condition basis.

[For this facility, acceptable and unacceptable combinations of out-of-service components are established as follows:]

a. For example, acceptable combinations of out-of-service components include:

   1. centrifugal charging pump in train A and SI pump in train B (components serve different functions),

   2. centrifugal charging pump in train A and containment sump isolation valve in train B (components are in separate, parallel flow paths);

b. For example unacceptable combinations of inoperable components include:

   1. centrifugal charging pump in train A and centrifugal charging control valve in train B failed closed (prevents charging flow to one cold leg), and

   2. SI cold leg check valve failed closed (example of a single component disabling part of both trains by preventing charging flow to one cold leg).

Reference 5 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train

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BASES (continued)

ACTIONS (continued)

is not available, the facility is in a condition outside the accident analysis. 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 placed in a MODE in which the LCO does not apply. This is done by placing the plant in MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience of the time required to reach the required plant conditions from full power in an orderly manner and without challenging plant systems.

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 they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type described in Reference 5 that can disable the function of both ECCS trains and invalidate the accident analyses. A 12-hour Frequency is considered reasonable in view of other administrative controls that will ensure that a mispositioned valve is unlikely.

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 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 non-accident position provided the valve will automatically reposition within the proper stroke time. This surveillance does not require any

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31-day Frequency of this SR was derived from inservice testing requirements for performing valve testing at least once every 92 days. The Frequency is further justified in view of the procedural controls governing valve operation and as a means providing added assurance of correct valve positions.

SR 3.5.2.3

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

SR 3.5.2.4

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 American Society of Mechanical Engineers (ASME) Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5 and SR 3.5.2.6

These surveillances demonstrate that each automatic ECCS valve actuates to its required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. The 18-month Frequency was developed considering it is prudent that these surveillances only be performed during a plant outage. This is due to the plant conditions needed to perform the SRs and the potential for unplanned plant transients if the SRs are 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 Engineered Safety Feature Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have stops to allow proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. This surveillance is not required for plants with flow-limiting orifices. The 18-month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18-month Frequency was developed considering it is prudent that this surveillance only be performed during a plant outage. This is due to plant conditions needed to perform the surveillance and access to the location. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

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BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling System."

2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants."

3. [Unit Name] FSAR, Section [ ], "[Emergency Core Cooling System]."

4. [Unit Name] FSAR, Section [15], "[Accident Analysis]."

5. IE Information Notice No. 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," January 6, 1987.

6. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5 3 ECCS—Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2 is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head), 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 refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. Included in these reductions is that automatic safety injection (SI) actuation is not available. Sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

An additional relaxation in the ECCS requirements for MODE 4 is that only one train of ECCS is required. This requirement dictates that single failures are not considered during this MODE of operation.

LCO

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

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BASES (continued)

LCO (continued)

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST 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 RWST 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 deliver its flow to the RCS hot and cold legs.

[For this facility, the following support systems are required to be OPERABLE to ensure ECCS RHR and high-head subsystem OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the ECCS RHR and high-head subsystems inoperable and their justification are as follows:]

APPLICABILITY

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

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "RHR and Coolant Circulation—High Water Level," and LCO 3.9.6, "RHR and Coolant Circulation—Low Water Level."

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BASES (continued)

ACTIONS

A.1

For this facility, an OPERABLE ECCS RHR subsystem consists of an RHR pump, a heat exchanger, piping, instruments, and controls to ensure an OPERABLE flow path.

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss-of-coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of 15 minutes to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay-heat removal is continuous.

With both RHR 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 RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

B.1

For this facility, an OPERABLE ECCS high-head subsystem consists of a centrifugal charging pump and flow path from RWST. The subsystem includes all the necessary piping, instruments, and controls required to ensure an OPERABLE flow path via centrifugal charging pump and RWST.

With no ECCS high-head subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high-pressure response to design basis events requiring SI. The 1-hour Completion Time to restore at least one ECCS high-head 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.

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BASES (continued)

ACTIONS (continued)

The Note associated with Required Action B.1 is intended to convey that continuation of actions is needed to restore the ECCS high-head subsystem to OPERABLE status, because the plant cannot be taken to MODE 5 since no decay heat removal capability is available.

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

The Note associated with Required Action C.1 is intended to restrict entry into this condition to when at least one RHR loop is OPERABLE. The Note also is intended to convey the suspension of further action to reach MODE 5 if, while in Condition C, all RHR loops become inoperable. Should the plant be in Condition A, no RHR loops OPERABLE, it is not advisable or practical to go to MODE 5. In this situation, the steam generators can be used to maintain MODE 4 until an RHR loop is restored to OPERABLE status. Should the plant be in Condition B only, an inoperable centrifugal charging loop, it is possible to reach MODE 5 by using an RHR loop.

SURVEILLANCE REQUIREMENTS

The applicable surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

The applicable references from Bases 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling, and to the ECCS and the Containment Spray System during accident conditions.

The RWST supplies both trains of the ECCS and the Containment Spray System through separate, redundant supply headers during the injection phase of a loss-of-coolant accident (LOCA) recovery. A motor-operated isolation valve is provided in each header to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump following receipt of the RWST—Low-Low (Level 1) signal. Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with the Design Basis Events.

The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS volume control tank (VCT) to the RWST through the use of isolation valves. Each set of isolation valves is interlocked so that the VCT isolation valves will begin to close once the RWST isolation valves are fully open. Since the VCT is under pressure, the preferred pump suction will be from the VCT until the tank is isolated. This will result in a delay in obtaining the RWST borated water. The effects of this delay are discussed in the Applicable Safety Analyses section of these Bases.

During normal operation in MODES 1, 2, and 3, the safety injection (SI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS and Containment Spray System pumps are provided with recirculation lines that ensure that each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

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BASES (continued)

BACKGROUND (continued)

When the suction for the ECCS and Containment Spray System pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that:

a. The RWST 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 System pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in insufficient cooling capacity of core spray (CS) when the transfer to the recirculation mode occurs. Improper boron concentrations could result in loss of SHUTDOWN MARGIN or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside containment.

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The importance of its value is small for units with a boron injection tank (BIT) with a high boron concentration. For units with no BIT or reduced BIT boron requirements, the minimum boron concentration limit is an important assumption in ensuring the required shutdown capability. The maximum

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BASES (continued)

BACKGROUND (continued)

boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a non-limiting event and the results are very insensitive to boron concentrations. The maximum temperature is a conservative assumption that minimizes the additional cooling in the feedline break event analysis; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically non-limiting.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory, and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of Bases B 3.5.2, "ECCS—Operating"; B 3.5.3, "ECCS—Shutdown"; and B 3.6.6, "Containment Spray and Cooling Systems." Reference to these analyses is used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as [27] seconds, with offsite power available, or [37] seconds without offsite power. This response time includes [2] seconds for electronics delay, a [15]-second stroke time for the RWST valves, and a [10]-second stroke time for the VCT valves. Plants with a BIT need not be concerned with the delay since the BIT will supply highly borated water prior to RWST switchover, provided the BIT is between the pumps and the core.

For a large-break LOCA analysis, the water volume limit of [466,200] gallons and the lower boron concentration limit of [2000] ppm are used to compute the post-LOCA sump boron

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

concentration necessary to assure subcriticality. The large-break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of [2200] ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of [35]°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of [100]°F is used in the small-break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small-break LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

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BASES (continued)

LCO (continued)

[For this facility, the following support systems are required to be OPERABLE to ensure RWST OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the RWST inoperable and their justification are as follows:]

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also 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." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal and Coolant Circulation—Low Water Level."

ACTIONS

A.1

If the RWST borated water volume, boron concentration, or borated water temperature is not within limits, it must be returned to within limits within 1 hour. In this condition, neither the ECCS nor the Containment Spray System can perform its design function. Under these conditions, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

If the equipment used to verify RWST borated water volume, temperature, or concentration is determined to be inoperable, the RWST is considered to be not within limits and Required Action A.1 applies to restore such equipment to OPERABLE status.

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BASES (continued)

ACTIONS (continued)

B.1 and B.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

Verification every [illegible] that the RWST borated water temperature is m[illegible] the limits assumed in the accident analy[illegible] is short enough 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 eliminates the requirement to perform this surveillance when ambient air temperatures are within the operating limits of the RWST. If ambient air temperatures are within the band, the RWST temperature should not exceed the limits.

[For this facility, RWST borated water temperature is measured as follows:]

SR 3.5.4.2

Verification every 7 days that the RWST water volume is maintained above the required minimum level will ensure that a sufficient initial supply is available for injection and for supporting continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7-day Frequency is appropriate and has been proven to be acceptable through operating experience.

[For this facility, RWST borated water volume is measured as follows:]

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.3

Verification every 7 days that the boron concentration of the RWST is within the required band ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7-day sampling Frequency to verify boron concentration is appropriate and has proven to be acceptable through operating experience.

[For this facility, RWST boron concentration is measured as follows:]

REFERENCES

1. [Unit Name] FSAR, Section [6], "[Title]," and Section [15], "[Title]."

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

This LCO is applicable only to those units that utilize the centrifugal charging pumps for safety injection (SI). The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during SI.

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large-break loss-of-coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small-break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

This LCO ensures that seal injection flow of ≤ [40] gpm, with centrifugal charging-pump discharge header pressure ≥ [2480] psig and charging flow control valve fully open, will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs,

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is determined by assuming that the RCS pressure is at normal operating pressure and that the centrifugal charging pump discharge pressure is greater than or equal to the value specified in this LCO. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the control valve (charging flow for four loop units and air-operated seal injection for three loop units) being fully open, is required since the valve is designed to fail open for the accident condition. With the discharge pressure and control valve position as specified by the LCO, a flow limit is established. It is this flow limit that is used in the accident analyses.

The limit on seal injection flow, combined with the centrifugal charging pump discharge header pressure limit and an open wide condition of the charging flow control valve, must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.

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BASES (continued)

LCO (continued)	[For this facility, the following support systems are required to be OPERABLE to ensure Seal Injection System OPERABILITY:] [For this facility, those required support systems which, upon their failure, do not require declaring the Seal Injection System inoperable and their justification are as follows:]
APPLICABILITY	In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.
ACTIONS	<u>A.1 and A.2</u> With the seal injection flow exceeding its limit, the amount of charging flow available to the RCS may be reduced. Under this condition, action must be taken to restore the flow to below its limit or to place the unit in a MODE in which this system is not required. The 1-hour Completion Time to restore the seal injection flow is for prompt action that will reduce the flow to within its limit. The adjustment to the flow can be made by either the manual valves or the charging flow control valve. If the initial adjustment to meet the Completion Time of 1 hour for Required Action A.1 is made with the charging flow control valve, the operator has 4 hours from the time the flow is known to be above the limit to correctly position the manual valves to meet the Completion Time of Required Action A.2 and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and ensures that seal injection flow is either restored to below its limit or that the plant is promptly placed in a MODE in which seal injection flow is not critical. These times are conservative with respect to

(continued)

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BASES (continued)

ACTIONS (continued)

the Completion Times of other ECCS LCOs, are based on operating experience, and are sufficient for taking corrective actions by operations personnel.

If equipment used to verify proper RCP seal injection flow is determined to be inoperable, the seal injection flow is considered to be not within limits and Required Actions A.1 and A.2 apply.

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a flow within the limit ensures that proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.

A provision has been added to except SR 3.0.4 for entry into MODE 3. The Note permits entry into MODE 3, since the SR cannot be performed in a lower MODE. The exception is permitted for up to 4 hours after the RCS pressure has stabilized within a $\pm$ 20 psig band about normal operating pressure. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the surveillance is timely.

(continued)

BASES (continued)

REFERENCES

1. [Unit Name] FSAR, Section [6], "[Title]," and Section [15], "[Title]."

2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Plants," 1974.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.6 Boron Injection Tank (BIT)

BASES

BACKGROUND

The BIT is part of the Boron Injection System, which is the primary means of quickly introducing negative reactivity into the Reactor Coolant System (RCS) on a safety injection (SI) signal.

The main flow path through the Boron Injection System is from the discharge of the centrifugal charging pumps through lines equipped with an flow element and two valves in parallel that open on an SI signal. The valves can be operated from the main control board. The valves and flow elements have main control board indications. Downstream of these valves, the flow enters the BIT (Ref. 1).

The BIT is a stainless steel tank containing concentrated boric acid. Two trains of strip heaters are mounted on the tank to keep the temperature of the boric acid solution above the precipitation point. The strip heaters are controlled by temperature elements located near the bottom of the BIT. The temperature elements also activate high and low alarms on the main control board. In addition to the strip heaters on the BIT, there is a recirculation system with a heat tracing system, including the piping section between the motor-operated isolation valves, which further ensures that the boric acid stays in solution. The BIT is also equipped with a high-pressure alarm on the main control board. The entire contents of the BIT are injected when required; thus, the contained and deliverable volumes are the same.

During normal operation, one of the two BIT recirculation pumps takes suction from the boron injection surge tank (BIST) and discharges to the BIT. The solution then returns to the BIST. Normally, one pump is running and one is shut off. On receipt of an SI signal, the running pump shuts off and the air-operated valves close. Flow to the BIT is then supplied from the centrifugal charging pumps. The solution of the BIT is injected into the RCS through the RCS cold legs.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

During a main steam line break (MSLB) or loss-of-coolant accident (LOCA), the BIT provides an immediate source of concentrated boric acid that quickly introduces negative reactivity into the RCS.

The contents of the BIT are not credited for core cooling or immediate boration in the LOCA analysis, but for post-LOCA recovery. The BIT maximum boron concentration of [22,500] ppm is used to determine the minimum time for hot leg recirculation switchover. The minimum boron concentration of [20,000] ppm is used to determine the minimum mixed mean sump boron concentration for post-LOCA shutdown requirements.

For the MSLB analysis, the BIT is the primary mechanism for injecting boron into the core to counteract any positive increases in reactivity caused by an RCS cooldown. The analysis uses the minimum boron concentration of the BIT, which also affects both the departure from nucleate boiling and containment design analyses. Reference to the LOCA and MSLB analyses is used to assess changes to the BIT to evaluate their effect on the acceptance limits contained in these analyses.

The minimum temperature limit of [145]°F for the BIT ensures that the solution does not reach the boric acid precipitation point. The temperature of the solution is monitored and alarmed on the main control board.

The BIT boron concentration limits are established to ensure that the core remains subcritical during post-LOCA recovery. The BIT will counteract any positive increases in reactivity caused by an RCS cooldown.

The BIT minimum water volume limit of [ ] gallons is used to ensure that the appropriate quantity of highly borated water with sufficient negative reactivity is injected into the RCS to shut down the core following an MSLB, to determine the hot leg recirculation switchover time, and to safeguard against boron precipitation.

The BIT satisfies Criteria 2 and 3 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

LCO

This LCO establishes the minimum requirements for contained volume, boron concentration, and temperature of the BIT inventory (Ref. 2). This ensures that an adequate supply of borated water is available in the event of a LOCA or MSLB to maintain the reactor subcritical following these accidents.

To be considered OPERABLE, the limits established in the SR for water volume, boron concentration, and temperature must be met.

If the equipment used to verify BIT parameters (temperature, volume, and boron concentration) is determined to be inoperable, then the BIT is also inoperable.

[For this facility, the following support systems are required to be OPERABLE to ensure BIT OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the BIT inoperable and their justification are as follows:]

APPLICABILITY

In MODES 1, 2, and 3, the BIT OPERABILITY requirements are consistent with those of LCO 3.5.2, "ECCS—Operating."

In MODES 4, 5, and 6, the respective accidents are less severe, so the BIT is not required in these lower MODES.

ACTIONS

A.1

If the required volume is not present in the BIT, both the hot leg recirculation switchover time analysis and the boron precipitation analysis would not be met. Under these conditions, prompt action must be taken to restore the volume to above its required limit to declare the tank OPERABLE, or the plant must be placed in a MODE in which the BIT is not required.

The BIT boron concentration is considered in the hot leg recirculation switchover time analysis, the boron precipitation analysis, and the reactivity analysis for an

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BASES (continued)

ACTIONS (continued)

MSLB. If the concentration were not within the required limits, these analyses could not be relied on. Under these conditions, prompt action must be taken to restore the concentration to within its required limits, or the plant must be placed in a MODE in which the BIT is not required.

The BIT temperature limit is established to ensure that the solution does not reach the boric acid crystallization point. If the temperature of the solution drops below the minimum, prompt action must be taken to raise the temperature and declare the tank OPERABLE, or the plant must be placed in a MODE in which the BIT is not required.

The 1-hour Completion Time to restore the BIT to OPERABLE status is consistent with other Completion Times established for loss of a safety function and assures that the plant will not operate long outside of the safety analyses.

B.1 and B.2

When Required Action A.1 cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power conditions and to be borated to the required SHUTDOWN MARGIN without challenging plant systems or operators. Borating to the required SHUTDOWN MARGIN assures that the plant is in a safe condition, without need for any additional boration.

C.1

After determining that the BIT is inoperable and the Required Actions of B.1 and B.2 have been completed, the tank must be returned to OPERABLE status within 7 days. These actions ensure that the plant will not be operated with an inoperable BIT for a lengthy period of time. It should be noted, however, that changes to applicable MODES cannot be made until the BIT is restored to OPERABLE status pursuant to the provisions of LCO 3.0.4.

D.1

Even though the RCS has been borated to a safe and stable condition as a result of Required Action B.2, either the

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BASES (continued)

ACTIONS (continued)

BIT must be restored to OPERABLE status (Required Action C.1) or the plant must be placed in a condition in which the BIT is not required (MODE 4). The 12 hour Completion Time to reach MODE 4 is reasonable, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.5.6.1

Verification every 24 hours that the BIT water temperature is at or above the specified minimum temperature is frequent enough to identify a temperature change that would approach the temperature limit. The solution temperature is also monitored by an alarm that provides further assurance of protection against low temperature. This Frequency has been shown to be acceptable through operating experience.

[For this facility, BIT borated water temperature is measured as follows:]

SR 3.5.6.2

Verification every 7 days that the BIT contained volume is above the required limit is frequent enough to assure that this volume will be available for quick injection into the RCS. If the volume is too low, the BIT would not provide enough borated water to ensure subcriticality during recirculation or to shut down the core following an MSLB. Since the BIT volume is normally stable, a 7-day Frequency is appropriate and has been shown to be acceptable through operating experience.

[For this facility, BIT borated water volume is measured as follows:]

SR 3.5.6.3

Verification every 7 days that the boron concentration of the BIT is within the required band ensures that the reactor remains subcritical following a LOCA, limits return to power following an MSLB, and maintains the resulting sump pH in an acceptable range so that boron precipitation will not occur

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

in the core. In addition, the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized.

The BIT is in a recirculation loop that provides continuous circulation of the boric acid solution through the BIT and the boric acid tank (BAT). There are a number of points along the recirculation loop where local samples can be taken. The actual location used to take a sample of the solution is specified in the plant surveillance procedures. Sampling from the BAT to verify the concentration of the BIT is not recommended, since this sample may not be homogenous and the boron concentration of the two tanks may differ. The sample should be taken from the BIT or from a point in the flow path of the BIT recirculation loop.

REFERENCES

1. [Unit Name] FSAR, Section [6], "[Title]," and Section [15], "[Title]."

2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Plants," 1974.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment (Ice Condenser)

BASES

BACKGROUND

The containment is a free-standing steel pressure vessel that is surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low-leakage steel shell that is designed to contain the radioactive material that may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and a concrete base mat with steel membrane. It is completely enclosed by a reinforced concrete shield building. An annular space exits between the walls and domes of the steel containment vessel and the concrete shield building to provide for the collection mixing, holdup, and controlled release of containment out leakage. Ice Condenser containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment integrity. The shield building provides biological shielding and allows controlled release of the annulus atmosphere under accident conditions, and environmental missile protection for the containment vessel and Nuclear Steam Supply System. The inner steel containment and its penetrations establish the leakage-limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission-product radioactivity from the containment to the environment. Loss of containment OPERABILITY could cause site-boundary doses, in the event of a DBA, to exceed values given in the

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BASES (continued)

BACKGROUND (continued)

licensing basis. All leakage-rate requirements and SRs are in conformance with 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions.

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, such that, in conjunction with the other containment systems and ENGINEERED SAFETY FEATURE systems, the release of fission-product radioactivity subsequent to a DBA will not result in doses in excess of the values given in the licensing basis.

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. 3). In addition, release of significant fission-product radioactivity within containment can occur from a LOCA or a REA. In the DBA analyses, it is assumed that the containment is OPERABLE at event initiation, 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 [0.1]% of containment air weight per day (Ref. 4). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 2), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. For this unit L_a = [0.1]% per day, and P_a = [14.4] psig, which results from the limiting design basis LOCA (Ref. 4).

Satisfactory leakage rate test results is a requirement for the establishment of containment OPERABILITY. The acceptance criteria applied to accidental releases of radioactive material to the environment are given in terms of total radiation dose received by:

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

a. A member of the general public who remains at the exclusion-area boundary for 2 hours following onset of the postulated fission-product release; or

b. A member of the general public who remains at the low-population-zone boundary for the duration of the accident.

The limits established in 10 CFR 100 (Ref. 1) are a whole-body dose of 25 rem, or a dose of 300 rem to the thyroid from iodine exposure, or both. The NRC staff-approved licensing basis, however, may use some fraction of these limits.

The containment satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The requirements stated in this LCO define the performance of the containment fission-product barrier. The containment design leakage rate (L_a) is an assumed initial condition. By limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J (Ref. 2), containment OPERABILITY is maintained.

The containment LCO requires that containment OPERABILITY be maintained. Other LCOs support this LCO by ensuring:

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

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

   2. closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Reference [ ];

b. All equipment hatches are closed;

c. Each airlock is OPERABLE (see LCO 3.6.2, Condition C, Note 1);

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BASES (continued)

LCO (continued)

d. The containment leakage rates are within their limits as defined in the Containment Leakage Rate Testing Program; and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

The Required Actions when other containment LCOs are not met have been specified in these LCOs and not in LCO 3.6.1.

Compliance with the LCO will ensure a containment configuration that is structurally sound and will limit leakage to those leakage rates assumed in the safety analysis. As a result, offsite radiation exposures will be maintained within the limits of 10 CFR 100 (Ref. 1) (or the NRC staff-approved licensing basis) following the most limiting DBA. The provisions of this LCO are implemented as follows:

a. OPERABILITY of containment penetrations:

1. The OPERABILITY of valves that are closed or are required to close in response to a containment isolation signal is guaranteed by compliance with the SRs of LCO 3.6.3, "Containment Isolation Valves." Some of the valves that must be closed to meet the accident analysis assumptions may be opened on an intermittent basis under administrative controls. These valves are identified in Reference [ ]. The Required Actions and SRs of LCO 3.6.3 ensure that the associated containment isolation valves either close within the required time limit, or the affected penetration is isolated by closed isolation valves or blind flanges, or the plant is shut down. In addition, the Type C tests required by SR 3.6.1.1 and Appendix J require that these containment isolation valves meet specified leakage rate criteria, namely, that the combined leakage rate for all penetrations and valves subject to Types B and C tests shall be less than 0.6 L_a.

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(continued)

BASES (continued)

LCO (continued)

2. The status of containment penetration isolation valves that are required to be closed during accident conditions, and that do not close automatically, is verified by SRs 3.6.3.1, 3.6.3.2, 3.6.3.3, and 3.6.3.4. The valves that must be closed to meet the accident analysis assumptions may be opened on an intermittent basis under administrative controls.

b. The OPERABILITY of the containment equipment hatch is assured by compliance with the leakage criteria established by 10 CFR 50, Appendix J (Ref. 2).

c. Containment air lock OPERABILITY is required by LCO 3.6.2, "Containment Air Locks," which requires that at least one door in each air lock be closed during MODES 1, 2, 3, and 4; that the air locks satisfy the required 10 CFR 50, Appendix J (Ref. 2), leakage test requirements, as described in the Containment Leakage Rate Testing Program; and that the door interlocks function as required.

d. Containment leakage-rate requirements are contained in 10 CFR 50, Appendix J (Ref. 2), and the Containment Leakage Rate Testing Program. These requirements are implemented to ensure that the reactor containment as a whole, and each of its penetrations and isolation valves, does not exceed the specified leakage rates.

e. The OPERABILITY of penetration sealing mechanisms is guaranteed by the successful completion of all the leakage testing requirements stipulated in 10 CFR 50, Appendix J (Ref. 2).

The measures implemented to meet the above requirements ensure that the containment will perform its designed safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 1) guidelines, or some fraction as established in the NRC staff-approved licensing basis.

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BASES (continued)

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 because of 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 refueling operations are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1-hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods where containment is inoperable is minimal.

B.1 and B.2

The plant must be placed in a MODE in which the LCO does not apply if containment cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant 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 MODES from full power in an orderly manner and without challenging plant systems.

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. 2), as modified by approved exemptions as described in the containment Leakage Rate Testing Program. This SR reflects the leakage-rate testing requirements with regard to overall containment leakage (Type A leakage tests); leakage from equipment hatch, electrical penetrations, and other penetrations (Type B leakage tests) except air locks; and containment

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

isolation valves (Type C leakage tests) except [42]-inch purge valves. Leakage-rate testing of the containment purge valves is addressed in LCO 3.6.3, "Containment Isolation Valves." Air-lock door-seal leakage testing is addressed in LCO 3.6.2, "Containment Air Locks." SR Frequencies are as required by Appendix J or identified in the Containment Leakage Rate Testing Program. Thus, SR 3.0.2 (which allows SR 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.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance."

2. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

3. [Unit Name] FSAR, Section [ ], "[Accident Analysis]."

4. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment (Atmospheric)

BASES

BACKGROUND

The containment is comprised of the concrete reactor building, 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) such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). 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. 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.

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 utilizing a three-way post-tensioning system.

The concrete reactor building 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. Loss of containment OPERABILITY could cause site-boundary doses, in the event of a DBA, to exceed values given in the licensing basis. All leakage-rate requirements and SRs conform with 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions.

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BASES (continued)

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 such that, in conjunction with the other containment systems and ENGINEERED SAFETY FEATURE systems, the release of fission-product radioactivity subsequent to a DBA will not result in doses in excess of the values given in the licensing basis.

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. 3). In addition, release of significant fission-product radioactivity within containment can occur from a LOCA or a REA. The DBA analyses assumed that the containment is OPERABLE at event initiation 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 [0.1]% of containment air weight per day (Ref. 4). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 2), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leak-rate testing. For this unit, L_a = [0.1]% per day and P_a = [44.1] psig, which results from the limiting design basis LOCA (Ref. 4).

Satisfactory leakage-rate test results is a requirement for the establishment of containment OPERABILITY. The acceptance criteria applied to accidental releases of radioactive material to the environment are given in terms of total radiation dose received by:

a. A member of the general public who remains at the exclusion-area boundary for 2 hours following onset of the postulated fission-product release; or

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BASFS (continued)

APPLICABLE SAFETY ANALYSES (continued)

b. A member of the general public who remains at the low-population-zone boundary for the duration of the accident.

The limits established in 10 CFR 100 (Ref. 1) are a whole-body dose of 25 rem or a dose of 300 rem to the thyroid from iodine exposure, or both. The NRC staff-approved licensing basis may use some fraction of these limits.

The containment satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The requirements stated in this LCO define the performance of the containment fission-product barrier. The containment design leakage rate (L_a) is an assumed initial condition. containment OPERABILITY is maintained by limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J (Ref. 2).

The c ta ment LCO requires that containment OPERABILITY be maint ed. Other containment LCOs support this LCO by ensuri.

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

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

2. closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Reference [ ];

b. All equipment hatches are closed;

c. Each air lock is OPERABLE (see LCO 3.6.2, Condition C, Note 1);

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BASES (continued)

LCO (continued)

d. The containment leakage rates are within their limits as defined in the Containment Leakage Rate Testing Program;

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE; and

f. The structural integrity of the containment is assured by the successful completion of the Containment Tendon Surveillance Program and by the associated visual inspections of the steel liner and penetrations for evidence of deterioration or breach of integrity.

The Required Actions when other containment LCOs are not met have been specified in those LCOs and not in LCO 3.6.1.

Compliance with the LCO will ensure a containment configuration that is structurally sound and will limit leakage to those leakage rates assumed in the safety analysis. As a result, offsite radiation exposures will be maintained within the limits of 10 CFR 100 (Ref. 1) (or the NRC staff-approved licensing basis) following the most limiting DBA. The provisions of this LCO are implemented as follows:

a. OPERABILITY of containment penetrations:

1. The OPERABILITY of valves that are closed or are required to close in response to a containment isolation signal is guaranteed by compliance with the SRs of LCO 3.6.3, "Containment Isolation Valves." Some of the valves that must be closed to meet the accident analysis assumptions may be opened on an intermittent basis under administrative controls. These valves are identified in Reference 4. The Required Actions and SRs of LCO 3.6.3 ensure that the associated containment isolation valves close within the required time limit, that the affected penetration is isolated by closed isolation valves or blind flanges, or that the plant is shut down. In addition, the Type C tests required by SR 3.6.1.1 and Appendix J require that these containment isolation valves meet specified leakage rate criteria, namely that the

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BASES (continued)

LCO (continued)

combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.6 L_a

2. The status of containment penetration isolation valves that are required to be closed during accident conditions, and do not close automatically, is verified by SR 3.6.3.1, SR 3.6.3.2, SR 3.6.3.3, and SR 3.6.3.4. The valves, which must be closed to meet the accident analysis assumptions, may be opened on an intermittent basis under administrative controls.

b. The OPERABILITY of the containment equipment hatch is assured by compliance with the leakage criteria established by 10 CFR 50 Appendix J (Ref. 2).

c. The OPERABILITY of containment air locks required by LCO 3.6.2, "Containment Air Locks," requires that at least one door in each air lock be closed during MODES 1, 2, 3, and 4; that the air locks satisfy the 10 CFR 50, Appendix J (Ref. 2), leakage test requirements, as described in the Containment Leakage Rate Testing Program; and that the door interlocks function as required.

d. The containment leakage rate requirements of 10 CFR 50, Appendix J (Ref. 2), and the Containment Leakage Rate Testing Program are implemented to ensure that the reactor containment as a whole, and each of its penetrations and isolation valves, does not exceed the specified leakage rates.

e. The successful completion of all the leakage-testing requirements stipulated in 10 CFR 50, Appendix J (Ref. 2), is necessary to ensure the OPERABILITY of penetration sealing mechanisms.

The measures implemented to meet the above requirements ensure that the containment will perform its designed safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 1) guidelines, or some fraction established in the NRC staff-approved licensing basis.

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BASES (continued)

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 refueling operations are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1-hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods where containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage-rate test requirements of 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions as described in the Containment Leakage Rate Testing Program. This SR reflects the

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

leakage-rate testing requirements with regard to overall containment leakage (Type A leakage tests); leakage from equipment hatch, electrical penetrations, and other penetrations (Type B leakage tests), except air locks; and containment isolation valves (Type C leakage tests), except [42]-inch purge valves. Leakage rate testing of the containment purge valves is addressed in LCO 3.6.3, "Containment Isolation Valves." Air-lock door-seal leakage testing is addressed in LCO 3.6.2, "Containment Air Locks." SR Frequencies are as required by Appendix J or identified in the Containment Leakage Rate Testing Program. Thus, SR 3.0.2 (which allows SR 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 containment with ungrouted, post-tensioned tendons, this surveillance ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 5).

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area Low Population Zone and Population Center Distance."

2. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

3. [Unit Name] FSAR, Section [ ], "Accident Analysis."

4. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

5. Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment (Subatmospheric)

BASES

BACKGROUND

The containment is comprised of the concrete reactor building, 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) such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). 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. 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.

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 utilizing a three-way post-tensioning system.

The concrete reactor building 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. Loss of containment OPERABILITY could cause site-boundary doses, in the event of a DBA, to exceed values given in the licensing basis. All leakage-rate and SRs conform with 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

the limiting DBA without exceeding the design leakage rate such that, in conjunction with the other containment systems and ENGINEERED SAFETY FEATURE systems, the release of fission-product radioactivity subsequent to a DBA will not result in doses in excess of the values given in the licensing basis.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss-of-coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref 2). In addition, release of significant fission-product radioactivity within containment can occur from a LOCA or REA. The DBA analyses assume that the containment is OPERABLE intact at event initiation 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 [0.1]% of containment air weight per day (Ref. 4). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 2), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage-rate testing. For this unit, L_a = [0.1]% per day and P_a = [40.4] psig, which results from the limiting design basis LOCA (Ref. 4).

Satisfactory leakage-rate test results is a requirement for the establishment of containment OPERABILITY. The acceptance criteria applied to accidental releases of fission-product radioactivity to the environment are given in terms of total radiation dose received by:

a. A member of the general public who remains at the exclusion-area boundary for 2 hours following onset of the postulated fission-product release; or

b. A member of the general public who remains at the low-population-zone boundary for the duration of the accident.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The limits established in 10 CFR 100 (Ref. 1) are a whole-body dose of 25 rem or a dose of 300 rem to the thyroid from iodine exposure, or both. The NRC staff-approved licensing basis may use some fraction of these limits.

The containment satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The requirements stated in this LCO define the performance of the containment fission-product barrier. The containment L_a is an assumed initial condition. containment OPERABILITY is maintained by limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J (Ref. 2).

The containment LCO requires that containment OPERABILITY be maintained. Other containment LCOs support this LCO by ensuring:

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

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

    2. closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Reference [ ];

b. All equipment hatches are closed;

c. Each air lock is OPERABLE (see LCO 3.6.2, Condition C, Note 1);

d. The containment leakage rates are within their limits as defined in the Containment Leakage Rate Testing Program;

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE; and

(continued)

(continued)

BASES (continued)

LCO (continued)

f. The structural integrity of the containment is assured by the successful completion of the Containment Tendon Surveillance Program and by the associated visual inspections of the steel liner and penetrations for evidence of deterioration or breach of integrity.

The Required Actions when other containment LCOs are not met have been specified in those LCOs and not in LCO 3.6.1.

Compliance with the LCO will ensure a containment configuration that is structurally sound and will limit leakage to those leakage rates assumed in the safety analysis. As a result, offsite radiation exposures will be maintained within the limits of 10 CFR 100 (Ref. 1) (or the NRC staff-approved licensing basis) following the most limiting DBA. The provisions of this LCO are implemented as follows:

a. OPERABILITY of containment penetrations:

1. The OPERABILITY of valves that are closed or required to close in response to a containment isolation signal is guaranteed by compliance with the SRs of LCO 3.6.3, "Containment Isolation Valves." Some of the valves that must be closed to meet the accident analysis assumptions may be opened on an intermittent basis under administrative controls. These valves are identified in Reference 4. The Required Actions and SRs of LCO 3.6.3 ensure that the associated containment isolation valves close within the required time limit, that the affected penetration is isolated by closed isolation valves or blind flanges, or that the plant is shut down. In addition, the Type C tests required by SR 3.6.1.1 and Appendix J require that these containment isolation valves meet specified leakage-rate criteria, namely that the combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.6 L_a

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BASES (continued)

LCO (continued)

2. The status of containment penetration isolation valves that are required to be closed during accident conditions, and do not close automatically, is verified by SRs 3.6.3.1, 3.6.3.2, 3.6.3.3, and 3.6.3.4. The valves, which must be closed to meet the accident analysis assumptions, may be opened on an intermittent basis under administrative controls.

b. The OPERABILITY of the containment equipment hatch is assured by compliance with the leakage criteria established by 10 CFR 50, Appendix J (Ref. 2).

c. The OPERABILITY of containment air locks required by LCO 3.6.2, "Containment Air Locks," requires that at least one door in each air lock be closed during MODES 1, 2, 3, and 4; that the air locks satisfy the required 10 CFR 50, Appendix J (Ref. 2), leakage-test requirements, as described in the Containment Leakage Rate Testing Program; and that the door interlocks function as required.

d. The containment leakage-rate requirements contained in 10 CFR 50, Appendix J (Ref. 2), and the Containment Leakage Rate Testing Program are implemented to ensure that the reactor containment as a whole, and each of its penetrations and isolation valves, does not exceed the specified leakage rates.

e. The OPERABILITY of penetration sealing mechanisms is guaranteed by the successful completion of all the leakage-testing requirements stipulated in 10 CFR 50, Appendix J (Ref. 2).

The measures implemented to meet the above requirements ensure that the containment will perform its designed safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 1) guidelines or some fraction established in the NRC staff-approved licensing basis.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6,

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BASES (continued)

APPLICABILITY (continued)

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 refueling operations are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1-hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods where containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage-rate test requirements of 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions as described in the Containment Leakage Rate Testing Program. This SR reflects the leakage-rate testing requirements with regard to overall containment leakage (Type A leakage tests); leakage from equipment hatch, electrical penetrations, and other penetrations except air locks (Type B leakage tests); and containment

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

isolation valves (Type C leakage tests), except [42]-inch purge valves. Leakage-REQUIREMENTS rate testing of the containment purge valves is addressed in LCO 3.6.3, "Containment Isolation Valves." Air-lock door-seal leakage testing is addressed in LCO 3.6.2, "containment Air Locks." SR Frequencies are as required by Appendix J or identified in the Containment Leakage Rate Testing Program. Thus, SR 3.0.2 (which allows SR 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 containments with ungrouted post-tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 5).

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance."

2. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

3. [Unit Name] FSAR, Section [ ] , "[Accident Analysis]."

4. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

5. Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment (Dual)

BASES

BACKGROUND

The containment is a free-standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low-leakage steel shell designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA) such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with a hemispherical dome and ellipsoidal bottom, completely enclosed by a reinforced concrete shield building. A 4-ft-wide annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to permit inservice inspection and collection of containment outleakage. Dual containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling and instrumentation pipelines into the containment vessel while maintaining containment OPERABILITY. The shield building provides biological shielding and allows controlled release of the annulus atmosphere under accident conditions as well as environmental missile protection for the containment vessel and the Nuclear Steam Supply System.

The inner steel containment and its penetrations establish the leakage-limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission-product radioactivity from the containment to the environment. Loss of containment OPERABILITY could cause site-boundary doses, in the event of a DBA, to exceed values given in the licensing basis. All leakage-rate requirements and SRs conform with 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions.

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BASES (continued)

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, such that, in conjunction with the other containment systems and ENGINEERED SAFETY FEATURE systems, the release of fission-product radioactivity subsequent to a DBA will not result in doses in excess of the values given in the licensing basis.

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. 3). In addition, release of significant fission-product radioactivity within containment can occur from a LOCA or a REA. The DBA analyses assume that the containment is OPERABLE at event initiation 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 [0.1]% of containment air weight per day (Ref. 4). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 2), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage-rate testing. For this unit, L_a = [0.1]% per day and P_a = [46.3] psig, which results from the limiting design basis LOCA (Ref. 4).

Satisfactory leakage-rate test results is a requirement for the establishment of containment OPERABILITY. The acceptance criteria applied to accidental releases of fission-product radioactivity to the environment are given in terms of total radiation dose received by:

a. A member of the general public who remains at the exclusion-area boundary for 2 hours following onset of the postulated fission-product release; or

b. A member of the general public who remains at the low-population-zone boundary for the duration of the accident.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The limits established in 10 CFR 100 (Ref. 1) are a whole-body dose of 25 rem or a dose of 300 rem to the thyroid from iodine exposure, or both. The NRC staff-approved licensing basis may use some fraction of these limits.

The containment satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The requirements stated in this LCO define the performance of the containment fission-product barrier. The containment design leakage rate (L_a) is an assumed initial condition. containment OPERABILITY is maintained by limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J (Ref. 2).

The containment LCO requires that containment OPERABILITY be maintained. Other containment LCOs support this LCO by ensuring:

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

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

2. closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Reference [ ];

b. All equipment hatches are closed;

c. Each air lock is OPERABLE (see LCO 3.6.2, Condition C, Note 1);

d. The containment leakage rates are within their limits as defined in the Containment Leakage Rate Testing Program; and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

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BASES (continued)

LCO (continued)

The Required Actions when other containment LCOs are not met have been specified in those LCOs and not in LCO 3.6.1.

Compliance with LCO 3.6.1 will ensure a containment configuration that is structurally sound and will limit leakage to those leakage rates assumed in the safety analysis. As a result, offsite radiation exposures will be maintained within the limits of 10 CFR 100 (Ref. 1) (or the NRC staff-approved licensing basis) following the most limiting DBA. The provisions of this LCO are implemented as follows:

a. OPERABILITY of containment penetrations:

1. The OPERABILITY of valves that are closed or are required to close in response to a containment isolation signal is guaranteed by compliance with the SRs of LCO 3.6.3, "Containment Isolation Valves." Some of the valves that must be closed to meet the accident analysis assumptions may be opened on an intermittent basis under administrative controls. These valves are identified in Reference 4. The Required Actions and SRs of LCO 3.6.3 ensure that the associated containment isolation valves close within the required time limit, that the affected penetration is isolated by closed isolation valves or blind flanges, or that the plant is shut down. In addition, the Type C tests required by SR 3.6.1.1 and Appendix J require that these containment isolation valves meet specified leakage-rate criteria, namely, that the combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.6 L_a.

2. The status of containment penetration isolation valves that are required to be closed during accident conditions, and do not close automatically, is verified by SRs 3.6.3.1, 3.6.3.2, 3.6.3.3, and 3.6.3.4. The valves that must be closed to meet the accident analysis assumptions may be opened on an intermittent basis under administrative controls.

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(continued)

BASES (continued)

LCO (continued)

b. The OPERABILITY of the containment equipment hatch is assured by compliance with the leakage criteria established by 10 CFR 50, Appendix J (Ref. 2).

c. The OPERABILITY of containment air locks, required by LCO 3.6.2, "Containment Air Locks," requires that at least one door in each air lock be closed during MODES 1, 2, 3, and 4; that the air locks satisfy the required 10 CFR 50, Appendix J (Ref. 2), leakage-test requirements, as described in the Containment Leakage Rate Testing Program; and that the door interlocks function as required.

d. Containment leakage-rate requirements are contained in 10 CFR 50, Appendix J (Ref. 2), and the Containment Leakage Rate Testing Program. These requirements are implemented to ensure that the reactor containment as a whole, and each of its penetrations and isolation valves, does not exceed the specified leakage rates.

e. The OPERABILITY of penetration sealing mechanisms is guaranteed by the successful completion of all the leakage-testing requirements stipulated in 10 CFR 50, Appendix J (Ref. 2).

The measures implemented to meet the above requirements ensure that the containment will perform its designed safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 1) guidelines, or some fraction established in the NRC staff-approved licensing basis.

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 refueling operations are addressed in LCO 3.9.4, "Containment Penetrations."

(continued)

BASES (continued)

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 that the probability of an accident (requiring containment OPERABILITY) occurring during periods where containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage-rate test requirements of 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions as described in the containment Leakage Rate Testing Program. This SR reflects the leakage-rate testing requirements with regard to overall containment leakage (Type A leakage tests); leakage from equipment hatch, electrical penetrations, and other penetrations (Type B leakage tests) except air locks; and containment isolation valves (Type C leakage tests) except [42]-inch purge valves. Leakage-rate testing of the containment purge valves is addressed in LCO 3.6.3, "Containment Isolation Valves." Air-lock door-seal leakage testing is addressed in LCO 3.6.2, "Containment Air Locks." SR Frequencies are as required by Appendix J or identified in the Containment Leakage Rate Testing Program. Thus, SR 3.0.2 (which allows SR 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.

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BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance."

2. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

3. [Unit Name] FSAR, Section [ ], "[Accident Analysis]."

4. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks (Atmospheric, Subatmospheric, Dual, and Ice Condenser)

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, 10 feet in diameter, with doors at each end, which are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air-lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door assures the containment is OPERABLE. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak-tight seal, the air lock design uses pressure-seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

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

The containment air locks form part of the containment pressure boundary. As such, air-lock integrity and air tightness is essential to limit offsite doses from a DBA. Not maintaining air-lock integrity or leak tightness may result in offsite doses in excess of those described in the plant safety analysis. All leakage-rate requirements and SRs are in conformance with 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The containment air lock LCO is derived from the requirements related to the control of offsite radiation does from major accidents by verifying that the actual containment leakage rate does not exceed the value assumed in the plant safety analysis. For example, the LOCA analysis requires the containment boundary to ensure that the site-boundary radiation dose will not exceed the Limits of 10 CFR 100 or the NRC staff-approved plant-specific licensing (e.g., specified fraction of 10 CFR 100 limits). As delineated in 10 CFR 100 (Ref. 2), the determination of exclusion areas and low-population zones surrounding a site must consider a fission-product release from the core with offsite release based on the expected demonstrable leakage rate from the containment.

The DBAs that result in a release of radioactive material within containment are a loss-of-coolant accident (LOCA), a steam line break, and rod ejection accident (REA) (Ref. 3). In the analysis of each of these accidents, it is assumed that containment is OPERABLE at event initiation, 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 [0.1]% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a [unit-specific #]: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) [unit-specific #] following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock. The acceptance criteria applied to DBA releases of radioactive material to the environment are given in terms of total radiation dose received by:

a. A member of the general public who remains at the exclusion-area boundary for 2 hours following onset of the postulated fission-product release; or

b. A member of the public who remains at the low-population-zone boundary for the duration of the accident.

The limits established in 10 CFR 100 are a whole-body dose of 25 rem or a dose of 300 rem to the thyroid from iodine

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

exposure, or both. The NRC staff-approved licensing basis may use some fraction of these limits.

Closure of a single door in each air lock is sufficient to ensure containment OPERABILITY following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry and exit from containment.

The containment air locks satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of offsite radiation exposures resulting from a DBA. Thus, each air lock's structural integrity and leakage 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. The closure of a single door in an air lock will maintain containment OPERABILITY, since each door is designed to withstand the peak containment pressure calculated to occur following a DBA.

This LCO provides assurance that the containment air locks will perform their designed safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the Reference 2 limits or some fraction thereof, as established by the NRC staff-approved licensing basis.

[For this facility, the following support systems are required to be OPERABLE to ensure containment air lock OPERABILITY:]

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BASES (continued)

LCO (continued)

[For this facility, those required support systems which, upon their failure, do not require declaring containment air locks inoperable and their justification are as follows:]

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 refueling operations are addressed in LCO 3.9.4, "Containment Building Penetrations."

The Required Actions of Conditions A, B, or C are modified by a Note that allows entry and exit to perform repairs on the affected air-lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through the outer 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 as low as reasonably achievable conditions permit, entry and exit should be via an OPERABLE air lock.

An additional Note has been added to provide clarification that for this LCO, all containment air locks are treated as an entity with a single Completion Time.

ACTIONS

A.1, A.2.1, A.2.2.1, and A.2.2.2

With one air-lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed, and must remain closed in each affected containment Air Lock. This assures a leak-tight containment barrier is maintained by the use of an OPERABLE air-lock door. This action must

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BASES (continued)

ACTIONS (continued)

be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires containment be restored to OPERABLE status within 1 hour.

In addition, the inoperable door in each affected air lock must be restored to OPERABLE status, or the affected air lock penetration must be isolated by locking closed the OPERABLE air-lock door. One of these two Required Actions must be completed within the 24 hour Completion Time. The 24-hour Completion Time is considered reasonable for restoring the air-lock door to OPERABLE status considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.2.2.2 verifies that an air lock with an inoperable door has been isolated by the use of a locked-closed OPERABLE air-lock door. This ensures that an acceptable containment leakage boundary is maintained. The leakage-rate acceptance criteria are as defined, in accordance with 10 CFR 50, Appendix J, within SR 3.6.2.1. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of other administrative controls, such as indications of door status available to the operator, which ensure that the OPERABLE air-lock door remains closed.

B.1, B.2.1, B.2.2.1, and B.2.2.2

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

Condition B is modified by a Note that allows entry and exit through an air lock under the control of a dedicated individual stationed at the air locks to ensure that only one door is opened at a time and that the opened door is immediately closed.

C.1 and C.2

With one or more air lock(s) inoperable for reasons other than those described in Condition A or B, one door in the containment Air Lock must be verified to be closed within a 1-hour Completion Time. This specified time period is

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BASES (continued)

ACTIONS (continued)

consistent with the ACTIONS of LCO 3. "Containment," which requires that containment be res ..ed to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be res to OPERABLE status within the 24-hour Completion Ti e specified time period is considered reasonable fo. ing inoperable air locks to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

The Required Actions of Condition C are modified by a Note that requires the containment to be declared inoperable should both doors in an air lock fail the air-lock door-seal test, SR 3.6.2.1.

D.1 and D.2

The plant must be placed in a MODE in which the LCO does not apply if the inoperable containment Air Lock cannot be restored to OPERABLE status within the associated Completion Times. This is done by placing the plant 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 MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage-rate test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions, and as described in the Containment Leakage Rate Testing Program. This SR reflects the leakage-rate testing requirements with regard to air-lock leakage (Type B leakage tests). The ceptance criteria are described in the unit Leakage Rate Test Program. The periodic testing requirements verify that the air-lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Surveillance Frequency is required by Appendix J, as modified by approved exemptions, and is described in the Containment Leakage Rate Testing Program. Thus, SR 3.0.2 (which allows Surveillance Frequency extensions) does not apply.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The SR has been modified by a Note to indicate an inoperable air-lock door does not invalidate the previous successful performance of an 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.

SR 3.6.2.2

The air-lock door interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post-accident containment pressure, closure of either door will ensure containment OPERABILITY. Thus, the door interlock feature ensures that containment OPERABILITY is maintained while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed, and that simultaneous inner and outer door opening will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when containment is entered, this test is only required to be performed prior to entering containment, but is not required more frequently than 184 days. The 184-day test interval is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

[For subatmospheric containments, the Frequency of this SR will be justified in the individual facility conversion to the new STS.]

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

2. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance."

3. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident-consequence-limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Locked-closed manual valves, deactivated 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. Closed systems are those systems designed in accordance with GDC 57 (Ref. 1). Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation (and possibly loss of containment OPERABILITY) or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system inside containment (in accordance with the requirements of 10 CFR 50, Appendix A, GDC 57). These barriers (typically containment isolation valves) make up the containment isolation system.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission-product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the purge and exhaust valves receive an isolation signal on a containment high radiation condition. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of

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BASES (continued)

BACKGROUND (continued)

fission-product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that adequate containment leak tightness is maintained during and after an accident by minimizing potential leakage paths to the environment. Therefore, the OPERABILITY requirements provide assurance that containment leakage rates assumed in the safety analysis will not be exceeded.

Shutdown Purge System ([42]-inch purge valves)

The Shutdown Purge System operates to supply outside air into the containment for ventilation and cooling or heating, and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the [42]-inch purge valves in some plants are not qualified for automatic closure from their open position under DBA conditions. Therefore, the [42]-inch purge valves are normally maintained closed in MODES 1 through 4 to ensure leak tightness.

Mini-Purge System ([8]-inch purge valves)

The Mini-Purge System operates to:

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

b. Equalize internal and external pressures.

Since the valves used in the Mini-Purge System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3 and 4.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the requirements related to the control of offsite radiation doses resulting from major accidents. As delineated in 10 CFR 100 (Ref. 2), the determination of exclusion areas

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

and low-population zones surrounding a proposed site must consider a fission-product release from the core with offsite release based upon the expected demonstrable leak rate from the containment. This LCO is intended to ensure that the offsite dose limits are not exceeded (i.e., that the actual containment leakage rate does not exceed the value assumed in the safety analysis). As part of the containment boundary, containment isolation valve OPERABILITY is essential to containment OPERABILITY. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

The DBAs that could result in a release of radioactive material within containment are a loss-of-coolant accident (LOCA) or 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 leakage paths to the environment through containment isolation valves (including containment purge valves) are minimized. The offsite dose calculations assumed that the [42]-inch purge valves were closed at event initiation. Likewise, it is assumed that the containment is isolated such that release of fission products to the environment is controlled by the rate of containment leakage.

The acceptance criteria applied to accidental releases of fission-product radioactivity to the environment are given in terms of total radiation dose received by:

a. A member of the general public who remains at the exclusion-area boundary for 2 hours following onset of the postulated fission-product release; or

b. A member of the general public who remains at the low-population-zone boundary for the duration of the accident.

The limits established in 10 CFR 100 (Ref. 2) are a whole-body dose of 25 rem or a dose of 300 rem to the thyroid from iodine exposure, or both. The NRC staff-approved licensing basis may use a specified fraction of these limits.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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 plant safety analysis was considered in the original design of the containment purge valves. Having two valves in series on each purge line provides assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with diverse power sources: motor-operated and pneumatically operated, spring closed, respectively. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.

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

The containment isolation valves (including containment purge valves) satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to control of offsite radiation exposures resulting from a DBA. This LCO addresses containment isolation valve OPERABILITY, stroke time, and containment purge valve leakage. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," under Type C testing.

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BASES (continued)

LCO (continued)

The automatic isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The containment purge valves have different OPERABILITY requirements. The [42]-inch purge valves must be maintained sealed-closed, and purge valves with resilient seals must meet additional leakage-rate requirements (SR 3.6.3.7). Also, purge system valves actuate on an automatic isolation signal. The valves covered by this LCO are listed along with their associated stroke times in the FSAR (Ref. 5).

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are locked-closed, automatic valves are deactivated and secured in their closed position (including check valves with flow through the valve secured), and blind flanges and closed systems are in place. Closed systems are those systems designed in accordance with GDC 57 (Ref. 1). These passive isolation valves/devices are those listed in Reference 3.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to mitigate the consequences of accidents that could result in offsite exposure comparable to the Reference 2 limits, or some fraction as established in the NRC-staff approved licensing basis.

[For this facility, the following support systems are required to be OPERABLE to ensure containment isolation valves OPERABILITY:]

[For this facility, those required support systems which upon their failure do not require declaring the containment isolation valves inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of containment isolation valves and the justification of whether or not each supported system is declared inoperable are as follows:]

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

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BASES (continued)

APPLICABILITY (continued)

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 and the containment purge valves are not required to be sealed-closed in MODE 5. The requirements for containment isolation valves and containment purge valves during MODE 6 refueling operations are addressed in LCO 3.9.4, "Containment Building Penetrations."

The Applicability is modified by a Note allowing normally locked- or sealed-closed containment isolation valves, except the [42]-inch purge valves, to be opened intermittently under administrative control. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the valve. In this way, the penetration can be rapidly isolated if a valid containment isolation signal 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, these valves may not be opened under administrative control. The provisions of LCO 3.0.4 apply.

A further Note has been added to provide clarification that each penetration flow path is independent and is treated as a separate entity with a separate Completion Time for the purposes of this LCO.

ACTIONS

A.1, A.2.1, A.2.2.1, and A.2.2.2

When one or more containment isolation valves is inoperable, at least one isolation valve must be verified to be OPERABLE in each affected open penetration. This action may be satisfied by examining logs or other information to determine if the valve is out of service for maintenance or other reasons. This Required Action is to be completed within 1 hour in order to provide assurance that a containment penetration is not open and causing a loss of containment OPERABILITY. The associated Completion Time is consistent with LCO 3.6.1, "Containment," and is considered a reasonable length of time to complete the Required Action.

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BASES (continued)

ACTIONS (continued)

In the event that one or more containment isolation valves are inoperable, either the inoperable valve must be restored to OPERABLE status or the affected penetration must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic containment isolation valve, a closed manual valve, a blind flange, or a check valve inside containment with flow through the valve secured. For penetrations isolated in accordance with Required Action A.2.2.1, the valve used to isolate the penetration should be the closest available one to containment. One of these two Required Actions must be completed within 4 hours. The 4-hour Completion Time is reasonable considering the time required to isolate the penetration and the relative importance of maintaining containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetrations that cannot be restored to OPERABLE status within the 4-hour Completion Time and have been isolated in accordance with Required Action A.2.2.1, the affected penetrations must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations that are required to be isolated following an accident and that are no longer capable of being automatically isolated, will be in the isolation position should an event occur. The Completion Time allowed for this is once every 31 days for valves outside containment and prior to entering MODE 4 from MODE 5 if not performed more often than once per 92 days for valves inside containment. The Completion Time of once per 31 days was developed based upon Inservice Inspection and Testing Program requirements to perform valve testing at least once per 92 days. This Required Action 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. For the valves inside containment, the time period specified as "prior to entering MODE 4 from MODE 5, if not performed more often than once per 92 days," is based on engineering judgment and is considered reasonable in view of the inaccessibility of the valves and other administrative controls that will ensure that valve misalignment is an unlikely possibility.

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BASES (continued)

ACTIONS (continued)

Condition A has been modified by a Note indicating that this Condition is not applicable to those penetrations with only one containment isolation valve and a closed system inside containment (i.e., the containment penetration is isolated in accordance with 10 CFR 50, Appendix A, GDC 57, Reference 1). The Required Actions for Condition A assume two valves in series are used to isolate the containment penetration and satisfy single-failure concerns.

Required Action A.1 has been further modified by a Note stating that Required Action A.1 is not applicable to penetrations that have only one isolation valve. Since the Note to Condition A excludes penetrations with only one isolation valve and a closed system inside containment, the Note to A.1 refers to penetrations with a single isolation valve on a system that is open inside containment but closed outside containment. For these systems, if the single isolation valve is inoperable, the intent is to go directly to Action A.2.1. These systems are very small piping lines, such as instrument lines which are closed systems outside of containment. The justification for a Completion Time of 4 hours is analogous to that for lines with two isolation valves. This Note only applies to small lines.

B.1, B.2.1, and B.2.2

When one or more containment isolation valves are inoperable, the inoperable valve(s) must be restored to OPERABLE status or the affected penetration must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, or a blind flange. A check valve may not be used to isolate the affected penetration, since GDC 57 (Ref. 1) does not consider the check valve an acceptable automatic isolation valve. One of these Required Actions 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 maintaining containment integrity during MODES 1, 2, 3, and 4. In the event that the affected penetration is isolated in accordance with Required Action B.2.1, the affected

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BASES (continued)

ACTIONS (continued)

penetration must be verified to be isolated on a periodic basis. This is necessary to ensure that containment OPERABILITY is maintained and that containment penetrations required to be isolated following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration is isolated is appropriate because the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating that this Condition is only applicable to penetrations with only one containment isolation valve and a closed system inside containment. This Note is necessary since this Condition is written to specifically address those penetrations isolated in accordance with 10 CFR 50, Appendix A, GDC 57 (Ref. 1). GDC 57 allows those lines that enter containment but are neither part of the reactor coolant pressure boundary nor connected directly to containment atmosphere to be isolated by means of one containment isolation valve.

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

In the event that one or more containment purge valves are not within the purge valve leakage limits, purge valve leakage must be restored to within limits or the affected penetration must be isolated. The method of isolation must use at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, or blind flange. One of these Required Actions must be completed within the 24-hour Completion Time. The specified time period is reasonable considering that the containment purge valves remain closed such that a gross breach of containment does not exist. For containment purge valves that are isolated in accordance with Required Action C.2.1, SR 3.6.3.7 must be performed at least once per 92 days. This ensures that degradation of the resilient seals is detected and confirms that the leakage rate of the containment purge valves does not increase during the time the penetration is isolated. The normal Frequency of SR 3.6.3.7 is 184 days and is based on an NRC initiative, Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 6). Since more reliance is being placed on a single valve during this condition, it is prudent to perform the SR more often.

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BASES (continued)

ACTIONS (continued)

Therefore, a periodic interval of once per 92 days is appropriate.

D.1

With one or more containment isolation valves inoperable in one or more penetration flow paths, verify that the Required Actions have been initiated for those supported systems declared inoperable by the support containment isolation valves within a Completion Time of [ ] hours.

The [ ]-hour Completion Time is defined as the most limiting of all the Required Actions for all the supported systems that need to be declared inoperable upon the failure of one or more support features specified under Condition D.

Required Action D.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of containment isolation valves have been initiated by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition D of this LCO.]

[For this facility, the identified supported systems Required Actions are as follows:]

E.1

With one or more containment isolation valves inoperable in one or more penetration flow paths, AND one or more required support or supported features, or both, inoperable associated with the other redundant penetration flow paths, the result is the loss of functional capability, and LCO 3.0.3 must be immediately entered. However, if the support or supported feature LCO, or both, takes into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

An example illustrating this situation would be when a support containment isolation valve is declared inoperable and subsequently is isolated in a penetration flow path associated with a supported ENGINEERED SAFETY FEATURE (ESF) system, then the other penetration flow paths associated with the redundant counterpart supported ESF systems and

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BASES (continued)

ACTIONS (continued)

their support systems must be OPERABLE; otherwise, a loss of functional capability exists. A loss of functional capability in this case may place the operation of the plant outside the safety analysis. Therefore, immediate action must be taken to bring the plant to a MODE outside the Applicability of the LCO for the containment isolation valves.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and at least MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each [42]-inch containment purge valve is required to be verified sealed-closed at 31-day intervals. This SR 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 prevent offsite dose limits from exceeding 10 CFR 100 limits (Ref. 2) or some fraction, as established in the NRC staff-approved licensing basis. Therefore, these valves are required to be in sealed-closed position during MODES 1, 2, 3, and 4. Containment purge valves that are sealed-closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The Surveillance Frequency is a result of an NRC initiative, Generic Issue B-24, related to containment purge valve use during plant operations (Ref. 7).

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2

This SR ensures that the [8]-inch purge valves are closed as required or, if open, open for an allowable reason. This SR has been modified by a Note indicating that these valves may be opened for pressure control, as low as reasonably achievable (ALARA) air quality considerations for personnel entry, and Surveillance tests that require the valve to be open. The [8]-inch purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31-day Surveillance Frequency is consistent with other containment isolation valve requirements discussed under SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that all containment isolation manual valves and blind flanges located outside containment and required to be closed during accident conditions are closed. The SR helps to ensure that post-accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. The Inservice Inspection and Testing Program requires valve testing on a 92-day Frequency. 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. Since verification of valve position for valves outside containment is relatively easy, the 31-day Frequency was chosen to provide added assurance of the correct positions.

Several Notes have been added to this SR. The first Note applies to valves and blind flanges located in high-radiation areas, and allows these valves to be verified as closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these valves once they have been verified to be in the proper position, is small. A second Note has been added that allows normally locked- or sealed-closed isolation valves to be opened intermittently under administrative controls. These

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the valve. In this way, the penetration can be rapidly isolated if a valid containment isolation signal is indicated. A third Note has been included to clarify that valves that are open under administrative controls are not required to meet the SR during the time the valves are open. The provisions of LCO 3.0.4 apply.

SR 3.6.3.4

This SR verifies that all containment isolation manual valves and blind flanges located inside containment and required to be closed during accident conditions are closed. The SR helps to ensure that post-accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For valves inside containment, the Frequency specified as "prior to entering MODE 4 from MODE 5 if not performed more often than once per 92 days" is appropriate since these valves and flanges are operated under administrative control and the probability of their misalignment is low.

A Note has been added to this SR that allows normally locked- or sealed-closed isolation valves to be opened intermittently under administrative controls. The administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the valve. In this way, the penetration can be rapidly isolated when a valid containment isolation signal is indicated. An additional Note has been included to clarify that valves that are open under administrative controls are not required to meet the SR during the time the valves are open. The provisions of LCO 3.0.4 apply.

SR 3.6.3.5

Demonstrating 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 that the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and Frequency of this SR are in

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

accordance with the Inservice Inspection and Testing Program, but the Frequency should not exceed 92 days.

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. The 18-month Frequency was developed considering it is prudent that this SR be performed only during a plant outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience has shown that these components usually pass this SR when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.7

For containment purge valves with resilient seals, additional leakage-rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 4), is required to ensure OPERABILITY. The individual purge valve leakage-rate limits for this unit are [ ]. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other types of seals. Based on this observation, and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 6).

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

A Note 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 containment purge valve leakage is properly accounted for in determining the overall containment leakage rate to verify containment OPERABILITY.

SR 3.6.3.8

In subatmospheric containments, the check valves that serve a containment isolation function are weight- or spring-loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.8 verifies the operation of the check valves that are testable during plant operation. The Frequency of 92 days is consistent with the Inservice Inspection and Testing Program requirement for valve testing on a 92-day Frequency.

SR 3.6.3.9

In subatmospheric containments, the check valves that serve a containment isolation function are weight- or spring-loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.9 verifies the operation of the check valves that are not testable during plant operation. The Frequency of 18 months is based on such factors as the inaccessibility of theses valves, the fact that the plant must be shut down to perform the tests, and the successful results of the tests on an 18-month basis during past plant operation.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants":

   General Design Criterion 50, "Containment Design Basis";

   General Design Criterion 52, "Capability for Containment Leakage Rate Testing";

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BASES (continued)

REFERENCES (continued)

General Design Criterion 53, "Provisions for Containment Inspection and Testing";

General Design Criterion 54, "Piping Systems Penetrating Containment";

General Design Criterion 56, "Primary Containment Isolation"; and

General Design Criterion 57, "Closed System Isolation Valves."

2. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance".
3. [Unit Name] FSAR, Section [ ], "[Accident Analysis]."
4. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."
5. [Unit Name] FSAR, Section [ ], "[Containment Systems]."
6. Generic Issue B-20 "Containment Leakage Due to Seal Deterioration."
7. Generic Issue B-24 "Containment Purge Valve Reliability."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4A Containment Pressure (Atmospheric, Dual, and Ice Condenser)

BASES

BACKGROUND

The containment serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA) such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). 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, a loss of containment OPERABILITY may result. In the event of a DBA, loss of containment OPERABILITY could cause site-boundary doses to exceed values specified in the licensing basis.

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 OPERABILITY, 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. 2).

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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 Containment Cooling System being inoperable). The containment analysis (Ref. 2) shows that the maximum peak calculated containment pressure, P_a, results from the limiting LOCA.

The initial pressure condition used in the containment analysis was [17.7] psia ([3.0] psig). The maximum containment pressure resulting from the worst-case LOCA, [44.1] psig, does not exceed the containment design pressure, [55] psig.

The containment was also designed for an external pressure load equivalent to [-2.5] psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was [-0.3] psig. This resulted in a minimum pressure inside containment of [-2.0] psig, which is less than the design load.

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 System 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. 3).

Containment pressure satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will

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BASES (continued)

LCO (continued)

remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative-differential pressure following the inadvertent actuation of the Containment Spray System. Maintaining containment pressure within the limits of this LCO ensures containment OPERABILITY.

[For this facility the following support systems are required to be OPERABLE to ensure containment pressure channel OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring containment pressure channels inoperable and their justification are as follows:]

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 containment OPERABILITY, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6 to ensure containment OPERABILITY.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1-hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

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BASES (continued)

ACTIONS (continued)

In the event that the required containment pressure channels are found inoperable, the containment pressure is considered to be not within limits and Required Action [A.1] applies.

B.1 and B.2

If containment pressure cannot be restored to within limits in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4A.1

Verifying that containment pressure is within limits ensures that facility operation remains within the limits assumed in the containment analysis. The 12-hour Frequency of this SR was developed based on operating experience related to trending of both containment pressure variations and pressure instrument drift during the applicable MODES and to assessing the proximity to the specified LCO pressure limits. Furthermore, the 12-hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."
2. [Unit Name] FSAR, Section [ ], "[Containment Analysis]."
3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4B Containment Pressure (Subatmospheric)

BASES

BACKGROUND

Containment air partial pressure is a process variable that is monitored and cont' olled. The containment air partial pressure is maintai . as a function of refueling water storage tank temperature and service water temperature according to Figure 3.6.4B-1 of the LCO, to ensure that, following a Design Basis Accident (DBA), the containment would depressurize in less than 60 minutes to subatmospheric conditions, and containment leakage would be such that offsite radiation exposures would be maintained within the requirements of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). Controlling containment partial pressure within prescribed limits also prevents the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of an inadvertent actuation of the Quench Spray (QS) System.

The containment internal air partial pressure limits of Figure 3.6.4B-1 are derived from the input conditions used in the containment DBA analyses. Limiting the containment internal air partial pressure and temperature in turn limits the pressure that could be expected following a DBA, thus ensuring containment OPERABILITY. Ensuring containment OPERABILITY limits leakage of fission-product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause si' -boundary doses, in the event of a DBA, to exceed values specified in the licensing basis.

APPLICABLE SAFETY ANALYSES

Containment air partial pressure is an initial condition used in the containment DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered relative to containment OPERABILITY are the loss-of-coolant accident (LOCA) and steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

to predict the resultant containment pressure transients. DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed assuming degraded containment ENGINEERED SAFETY FEATURE (ESF) systems (i.e., assuming the loss of one ESF bus), which is the worst-case single active failure, resulting in one train of the QS System and one train of the Recirculation Spray System becoming inoperable. The containment analysis for the DBA (Ref. 2) shows that the maximum peak containment pressure, P_a, results from the limiting design basis LOCA.

The maximum design internal pressure for the containment is [4 .0] psig. The initial conditions used in the containment design basis analyses were an air partial pressure of [12.2] psia and an air temperature of [120]°F. This resulted in a maximum peak containment internal pressure of [44.9] psig, which is less than the maximum design internal pressure for the containment.

The containment was also designed for an external pressure load of [9.2] psid (i.e., a design minimum pressure of [5.5] psia). The inadvertent actuation of the QS System was analyzed to determine the reduction in containment pressure (Ref. 2). The initial conditions used in the analysis were [8.6] psia and [120]°F. This resulted in a minimum pressure inside containment of [7.7] psia, which is considerably above the design minimum of [5.5] psia.

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 System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For the reflood phase calculations, 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. 3).

Containment pressure satisfies Criterion 2 of the NRC Interim Policy Statement.

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BASES (continued)

LCO

Maintaining containment pressure within the limits shown in Figure 3.6.4B-1 of the LCO ensures that in the event of a DBA the resultant peak containment accident pressure will be maintained below the containment design pressure. 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 QS System. With containment pressure maintained within the limits of this LCO, containment OPERABILITY is ensured. The LCO limits also ensure the return to subatmospheric conditions within 60 minutes following a DBA.

[For this facility, the following support systems are required to be OPERABLE to ensure containment pressure channel OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring containment pressure channels inoperable and their justification are as follows:]

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 containment OPERABILITY, 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 Reactor Coolant System pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6 to ensure containment OPERABILITY.

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

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BASES (continued)

ACTIONS (continued)

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 in 1 hour.

In the event that the required containment pressure channels are found inoperable, the containment pressure is considered to be not within limits, and Required Action A.1 applies.

B.1 and B.2

If containment pressure cannot be restored in the required time period, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4B.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 considering operating experience related to trending of both containment pressure variations and pressure instrument drift during the applicable MODES and to assessing the proximity to the specified LCO pressure limits. Furthermore, the 12-hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. [Unit Name] FSAR, Section [ ], "[Containment Analysis]."

3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5A Containment Air Temperature (Atmospheric & Dual)

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the guidelines of 10 CFR 100 (Ref. 1), or the NRC staff-approved licensing basis (e.g., specified fraction of 10 CFR 100 limits). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss-of-coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during plant operations. The total amount of energy to be removed from containment by the containment spray and cooling systems during post-accident conditions is dependent upon the quantity of energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. THE DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with respect to ENGINEERED SAFETY FEATURE (ESF) systems, assuming the loss of one ESF bus, which is the worst-case single active failure, resulting in one train each of the Containment Spray System, Residual Heat Removal System, and Containment Cooling System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is an SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 2) is [120]°F. This resulted in a maximum containment air temperature of [384.9]°F. The design temperature is [320]°F.

The temperature limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature for [unit-specific time period] during the transient. The basis of the containment design temperature, however, is to ensure the OPERABILITY of safety-related equipment inside containment (Ref. 3). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA SLB.

The temperature limit is also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System (Ref. 2).

The containment pressure transient is sensitive to the initial air mass in containment and therefore the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is the LOCA. The temperature limit is used in this analysis

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

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

LCO

During a DBA, with an initial containment average temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is computed to remain within acceptable limits. As a result, the ability of containment to perform its design function is ensured.

[For this facility, the following support systems are required to be OPERABLE to ensure containment air temperature channel OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the containment air temperature channel inoperable and their justification are as follows:]

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

With containment average air temperature outside the subject LCO limit, the average air temperature must be restored to within its limit within 8 hours. This action must be taken to return the unit 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 or to prepare the plant for an orderly shutdown.

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BASES (continued)

ACTIONS (continued)

In the event that the required containment air temperature channels are found inoperable, the containment air temperature is considered to be not within limits and Required Action [A.1] applies.

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The plant must be placed in a MODE in which the LCO does not apply if the containment average air temperature cannot be restored to within its limit within the associated Completion Time. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowable Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5A.1

Verifying that the containment average air temperature is within the LCO limit ensures that containment operation remains within the limits assumed in 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 that were selected to be representative of the overall containment atmosphere. The 24-hour Frequency of this SR was developed considering operating experience related to containment temperature variations and temperature instrument drift during the applicable MODES. Furthermore, the 24-hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

3. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5B Containment Air Temperature (Ice Condenser)

BASES

BACKGROUND

The containment structure serves to contain radioactive material which may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., specified fraction of 10 CFR 100 limits). The containment average air temperature is limited, during normal operation, to preserve the initial conditions assumed in the accident analyses for a loss-of-coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during plant operations. The total amount of energy to be removed from containment by the containment spray and cooling systems during post-accident conditions, is dependent upon the quantity of energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in a higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment. The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously, or consecutively. The postulated DBAs are analyzed in regard to ENGINEERED SAFETY FEATURE (ESF) systems, assuming the loss of one ESF bus, which is the worst-case single active failure, resulting in one train each of Containment Spray System, Residual Heat Removal System, and Air Return System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is an SLB. For the upper compartment, the initial containment average air temperature assumed in the design basis analyses (Ref. 2) is [110]°F. For the lower compartment, the initial average containment air temperature assumed in the design basis analyses is [120]°F. This resulted in a maximum containment air temperature of [326]°F. The design temperature is [250]°F.

The temperature upper limits are used to establish the environmental qualification operating envelope for both containment compartments. The maximum peak containment air temperature for both containment compartments was calculated to exceed the containment design temperature for a [unit-specific time period] during the transient. The basis of the containment design temperature, however, is to ensure the OPERABILITY of safety-related equipment inside containment (Ref. 3). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperatures are acceptable for the DBA SLB.

The temperature upper limits are also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System (Ref. 4) for both containment compartments.

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

is the LOCA. The temperature lower limits, [85]°F for the upper compartment and [100]°F for the lower compartment, are used in this analyses to ensure that, in the event of an accident, the maximum containment internal pressure will not be exceeded in either containment compartment.

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

LCO

During a DBA, with an initial containment average temperature within the LCO temperature limits, the resultant peak accident temperature is calculated to remain within acceptable limits. As a result, the ability of containment to perform its design function is ensured. In MODES 3 and 4, containment air temperature may be as low as 60°F, because the resultant calculated peak containment accident pressure would not exceed the design pressure due to a lesser amount of energy released from the pipe break in these MODES.

[For this facility, the following support systems are required to be OPERABLE to ensure containment air temperature OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the containment air temperature inoperable and their justification are as follows:]

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.

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BASES (continued)

ACTIONS

A.1

With containment average air temperature in the upper or lower compartment outside the subject LCO limits, the average air temperature in the affected compartment must be restored to within its limits within 8 hours. This action must be taken to return the unit 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 or to prepare the plant for an orderly shutdown.

In the event that the required containment air temperature channels are found inoperable, the containment air temperature is considered to be not within limits and Required Action [A.1] applies.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limits within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowable Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5B.1 and SR 3.6.5B.2

Verifying that the containment average air temperature is within the LCO limits ensures that containment operation remains within the limits assumed in the containment analyses. In order to determine the containment average air temperature, a weighted average is calculated using measurements taken at locations within the containment that were selected to be representative of the overall containment atmosphere. The 24-hour Frequency of these SRs was developed considering operating experience related to containment temperature variations and temperature instrument drift during the applicable MODES.

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Furthermore, the 24-hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

3. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5C <u>Containment Air Temperature</u> (Subatmospheric)

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the guidelines of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., specified fraction of 10 CFR 100 limits). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss-of-coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during plant operations. The total amount of energy to be removed from containment by the containment spray and cooling systems during post-accident conditions is dependent upon the quantity of energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy which must be removed, resulting in a higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed in regards to containment ENGINEERED SAFETY FEATURES (ESF) systems, assuming the loss of one ESF bus, which is the worst-case single active failure, resulting in one train of the Quench Spray (QS) system, and Recirculation Spray System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is an SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 2) is [120]°F. This resulted in a maximum containment air temperature of [357]°F. The design temperature is [347]°F.

The temperature upper limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature for a [unit specific time period] during the transient. The basis of the containment design temperature, however, is to ensure the OPERABILITY of safety-related equipment inside containment (Ref. 3). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA SLB.

The temperature upper limit is also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the QS System (Ref. 2).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is the LOCA. The temperature lower limit is used in this analyses to ensure that, in the event of an accident, the maximum containment internal pressure will not be exceeded.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

During a DBA, with an initial containment average temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is computed to remain within acceptable limits. As a result, the ability of containment to perform its design function is ensured.

[For this facility, the following support systems are required to be OPERABLE to ensure the containment air temperature channels OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring containment air temperature channels inoperable and their justification are as follows:]

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

With containment average air temperature outside the subject LCO limits, the average air temperature must be restored to within its limits within the 8 hours. This action must be taken to return the unit 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 or to prepare the plant for an orderly shutdown.

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BASES (continued)

ACTIONS (continued)

In the event that the required containment air temperature channels are found inoperable, the containment air temperature is considered to be not within limits and Required Action [A.1] applies.

B.1 and B.2

The plant must be placed in a MODE in which the LCO does not apply if the containment average air temperature cannot be restored to within its limits within the associated Completion Time. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowable Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5C.1

Verifying that the containment average temperature is within the LCO limits ensures that containment operation remains within the limits assumed in the containment analyses. In order to determine the containment average air temperature, a weighted average is calculated using measurements taken at locations within containment that were selected to be representative of the overall containment atmosphere. The 24-hour Frequency of this SR was developed considering operating experience related to containment temperature variations and temperature instrument drift during the applicable MODES. Furthermore, the 24-hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Center Distance."
2. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

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BASES (continued)

REFERENCES (continued)	3.	Title 10, Code of Federal Regulation, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6A Containment Spray and Cooling Systems (Atmospheric and Dual)
(Credit taken for iodine removal by Containment Spray System)

BASES

BACKGROUND

Containment Spray System

The Containment Spray System supports containment OPERABILITY by furnishing 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 less than the guidelines of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., the specified fraction of 10 CFR 100 limits). The containment spray and cooling systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," and GDC 40, "Testing of Containment Heat Removal Systems" (Ref 2), or other documents that were appropriate at the time of licensing (identified on a plant-specified basis).

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

The Containment Spray System provides a spray of cold borated water into the upper regions of Containment to reduce the Containment pressure and temperature during a DBA. The RWST solution temperature is an important factor in determining the heat-removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the residual heat removal coolers. Each train of the Containment Spray System provides adequate

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BASES (continued)

BACKGROUND (continued)

spray coverage to meet the system design requirements for containment heat removal.

The Spray Additive System injects a sodium hydroxide (NaOH) solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically, by a containment High-3 pressure signal, or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two containment spray pumps, and begins the injection phase. 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. The injection phase continues until a RWST level Low-Low alarm is received. The Low-Low level alarm for the RWST actuates valves to align the containment spray system pump suction with the containment sump and/or signals the operator to manually align the system to recirculation mode. The containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

Containment Cooling System

The Containment Cooling System is designed to furnish normal containment atmosphere cooling and to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure, in conjunction with the iodine-removal capability of the Containment Spray System, reduces the release of fission-product radioactivity from containment to the environment, in the event of a DBA, to less than the guidelines in the licensing basis.

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BASES (continued)

BACKGROUND (continued)

Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. Each train of two fan units is supplied with cooling water from a separate train of essential service Water (ESW). Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, and instrument tunnel, and outside the secondary shield in the lower areas of containment.

During normal operation all four fan units are operating. The fans are normally operated at high speed with ESW supplied to the cooling coils. The Containment Cooling System, operating in conjunction with the Containment ventilation and air-conditioning systems, is designed to limit the ambient containment air temperature during normal plant operation to less than the limit specified in LCO 3.6.5A, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

In post-accident operation, following an actuation signal the Containment Cooling System, fans are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere. The temperature of the ESW is an important factor in the heat-removal capability of the fan units.

The Containment Cooling System and Containment Spray System are 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 the Containment Cooling System provide redundant methods to limit and maintain post-accident conditions to less than the containment design values.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System protect the integrity of the containment by limiting the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

integrity are the loss-of-coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant Containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst-case single active failure and results in one train of the Containment Spray System and Containment Cooling System being rendered inoperable.

The analysis and evaluation show that under the worst-case scenario, the highest peak containment pressure is [44.1] psig (experienced during a LOCA). The analysis shows that the peak containment temperature is [384.5]°F (experienced during an SLB). Both results meet the intent of the design basis. (See Bases B 3.6.4A, "Containment Pressure," and B 3.6.5A, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a unit-specific power level of [ ], one containment spray train and one containment cooling train operating, and initial (pre-accident) containment conditions of [120]°F and [1.5] psig. The analyses also assume a response-time-delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System (ECCS) during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a 2.0 psig containment pressure and is associated with the sudden cooling effect in the interior of the air-tight containment. Additional discussion is provided in Bases 3.6.4A, "Containment Pressure."

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-3 pressure setpoint to achieve full flow through the Containment spray nozzles. The Containment Spray System total response time of [60] seconds includes diesel generator startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray-line filling (Ref. 4).

Containment cooling train performance for post-accident conditions is given in Reference 5. The result of the analysis is that each train can provide 100% 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 6.

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-3 pressure setpoint to achieving full Containment Cooling System air and safety-grade cooling water flow. The Containment Cooling System total response time of [60] seconds, includes signal delay, diesel generator startup (for loss of offsite power), and service water pump startup times (Ref. 7).

The containment spray and cooling systems satisfy Criterion 3 of the NRC Interim Policy Statement.

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 (Ref. 8). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain offsite doses below the guidelines of the licensing basis. To ensure that these requirements are met, two containment spray trains and two Containment cooling trains must be OPERABLE. Therefore, in the event of an accident, at least one train in each system operates, assuming the worst-case single active failure occurs.

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BASES (continued)

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 RWST upon an ESF actuation signal and automatically transferring suction to the containment sump.

Each Containment Cooling System typically includes demisters, cooling coils, dampers, fans, instruments and controls to ensure an OPERABLE flow path.

[For this facility, an OPERABLE Containment Cooling System constitutes the following:]

[ ]

[For this facility, the support systems required to be OPERABLE to ensure Containment Spray System and Containment Cooling System OPERABILITY are as follows:]

[ ]

[For this facility, those required support systems which upon their failure do not require declaring the containment spray and cooling systems inoperable and their justification are as follows:]

In addition, each Containment Spray System and Containment Cooling System must satisfy all the performance and physical arrangement requirements set forth by the SRs in order to be considered OPERABLE.

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.

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BASES (continued)

APPLICABILITY (continued)

In MODE 3 or MODE 4, individual plants may justify removal of the Containment Spray System from operation to support Shutdown Cooling System operation. In this condition, the Containment Cooling System must remain OPERABLE. Justification of Containment Spray System removal will be addressed on a plant-specific basis.

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.

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 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 low probability of a DBA occurring during this period.

B.1 and B.2

The plant must be placed in a MODE in which the LCO does not apply if the inoperable Containment Spray System cannot be restored to OPERABLE status within the associated Completion Time. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 within 84 hours. The 6 hours allotted to reach MODE 3 is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the Containment Spray System in MODE 3, and 36 hours to reach MODE 5. This is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

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BASES (continued)

ACTIONS (continued)

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 greater than 100% of the heat-removal needs (for the Condition with one containment cooling train inoperable). 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 DBA occurring during this period.

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 provide iodine-removal capabilities, and are capable of providing greater than 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, the iodine-removal function of the Containment Spray System, and the low probability of DBA occurring during this period.

E.1

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

F.1 and F.2

The plant must be placed in a MODE in which the LCO does not apply if the Required Actions and associated Completion Times for Condition C or D of this LCO are not met. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable time, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.6A.1

Verifying the correct alignment for manual, power-operated, and automatic valves, in the Containment Spray System flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct positions prior to locking, sealing, or securing. The 31-day Frequency of this SR was developed based upon Inservice Inspection and Testing Program requirements to perform valve testing at least once per 92 days. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (only check valves are inside containment) and capable of potentially being mispositioned are in the correct position.

SR 3.6.6A.2

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

SR 3.6.6A.3

Verifying a containment cooling train ESW cooling flow rate of ≥ [700] gpm to each cooling unit provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 4). The 31-day Frequency of this SR was developed based on Inservice Inspection and Test Program requirements to perform testing on safety-related components at least once per 92 days. The Frequency was also 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6A.4

Demonstrating each containment spray pump develops ≥ [ ] psid differential pressure on recirculation ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the American Society of Mechanical Engineers (ASME) Code (Ref. 9). Since the containment spray 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 inspections confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Inspection and Testing Program.

SR 3.6.6A.5 and SR 3.6.6A.6

These SRs require demonstration that each automatic containment spray valve actuates to its correct position and each containment spray pump starts on receipt of an actual or simulated actuation of a containment High-3 pressure signal. The 18-month Frequency was developed considering it is prudent that these Surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unplanned plant transients if the SR is performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The Surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single Surveillance may be used to satisfy both requirements.

SR 3.6.6A.7

This SR requires a demonstration that each containment cooling unit actuates upon receipt of an actual or simulated safety injection signal. The 18-month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See the SR 3.6.6A.5 and

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6A.6, above, for further discussion on the basis for the 18-month Frequency.

SR 3.6.6A.8

With the containment spray inlet valves closed and the spray header drained of any solution, low-pressure air or smoke can be blown through test connections. This SR ensures 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 design of the nozzle, a test at the first refueling and then at 10-year intervals is considered adequate to detect degradation in the performance of the nozzles.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Model."

4. [Unit Name] FSAR, Section [ ], "[Title]."

5. [Unit Name] FSAR, Section [ ], "[Title]."

6. [Unit Name] FSAR, Section [ ], "[Title]."

7. [Unit Name] FSAR, Section [ ], "[Title]."

8. [Unit Name] FSAR, Section [ ], "[Title]."

9. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6B Containment Spray and Cooling Systems (Atmospheric & Dual)
(Credit not taken for iodine removal by Containment Spray System)

BASES

BACKGROUND

Containment Spray System

The Containment Spray System supports containment OPERABILITY by furnishing containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission-product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to less than the guidelines of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The Containment Spray and Cooling Systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," and GDC 40, "Testing of Containment Heat Removal Systems" (Ref. 2), or other documents that were appropriate at the time of licensing (identified on a plant-specific basis).

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

The Containment Spray System provides a spray of cold borated water into the upper regions of Containment to reduce the Containment pressure and temperature during a DBA. The RWST solution temperature is an important factor in determining the heat-removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the residual heat removal coolers. Each train of the Containment Spray System provides adequate

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BASES (continued)

BACKGROUND (continued)

spray coverage to meet the system design requirements for containment heat removal.

The Containment Spray System is actuated either automatically, by a containment High-3 pressure signal, or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two containment spray pumps, and begins the injection phase. 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. The injection phase continues until an RWST level Low-Low alarm is received. The Low-Low level alarm for the RWST actuates valves to align the containment spray pump suction to the containment and/or sump signals the operator to manually align the system to recirculation mode. The Containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

Containment Cooling System

The Containment Cooling System is designed to furnish normal containment atmosphere cooling and to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission-product radioactivity from containment to the environment, in the event of a DBA, to less than the guidelines in the licensing basis.

Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirements, are provided. Each train of two fan units is supplied with cooling water from a separate train of essential service water (ESW). Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, instrument tunnel, and outside the secondary shield in the lower areas of containment.

During normal operation, four fan units are operating. The fans are normally operated at high speed, with ESW supplied to the cooling coils. The Containment Cooling System,

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BASES (continued)

BACKGROUND (continued)

operating in conjunction with the containment ventilation and air-conditioning systems, is designed to limit the ambient containment air temperature during normal plant operation to less than the limit specified in LCO 3.6.5A, "Containment Air Temperature." This temperature limitation, ensures that the Containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

In post-accident operation, following an actuation signal, the Containment Cooling System fans are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher-mass atmosphere. The temperature of the ESW cooling is an important factor in the heat-removal capability of the fan units.

The Containment Cooling System and Containment Spray System are 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 the Containment Cooling System provide redundant methods to limit and maintain post-accident conditions to less than the containment design values.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System ensure containment OPERABILITY by limiting the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to Containment integrity are the loss-of-coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant Containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF Systems, assuming the loss of one ESF bus, which is the worst-case single active failure and results in one train of Containment Spray System and Containment Cooling System being inoperable.

The analysis and evaluation show that, under the worst-case scenario, the highest peak Containment pressure is 44.1 psig (experienced during a LOCA). The analysis shows that the

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

peak containment temperature is [384]°F (experienced during an SLB). Both results meet the intent of the design basis. (See Bases B. 3.6.4A, "Containment Pressure," and B 3.6.5A, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a unit-specific power level of [ ], one containment spray train and one containment cooling train operating, and initial (pre-accident) containment conditions of [120]°F and [1.5] psig. The analyses also assume a response-time-delayed initiation in order to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System (ECCS) during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a [-2.0] psig containment pressure and is associated with the sudden cooling effect in the interior of the air-tight containment. Additional discussion is provided in Basis B 3.6.4A, "Containment Pressure."

The modeled Containment Spray System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-3 pressure setpoint to achieve full flow though the containment spray nozzles. The Containment Spray System total response time of [60] seconds includes diesel generator startup (for loss of offsite power), block-loading of equipment, containment spray pump startup, and spray-line filling (Ref. 4).

Containment cooling train performance for post-accident conditions is given in Reference 5. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post-accident condition. The train post-accident cooling capacity under varying

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

containment ambient conditions, required to perform the accident analyses, is also shown in Reference 6.

The modeled Containment Cooling System actuation from the Containment analysis is based on a response time associated with exceeding the containment High-3 pressure setpoint to achieve full Containment Cooling System air and safety-grade cooling water flow. The Containment Cooling System total response time of [60] seconds includes signal delay, diesel generator startup (for loss of offsite power), and Service Water pump startup times (Ref. 7).

The Containment Spray System and Containment Cooling System satisfy Criterion 3 of the NRC Interim Policy Statement.

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 (Ref. 8). 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, at least one train in each system operates, assuming the worst-case single active failure occurs.

Each Containment Spray System typically includes a spray pump, spray headers, nozzles, valves, piping, instruments and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and automatically transferring suction to the containment sump.

Each Containment Cooling System typically includes demisters, cooling coils, dampers, instruments, and controls to ensure an OPERABLE flow path.

[For this facility, an OPERABLE Containment Spray System and an OPERABLE Containment Cooling System constitutes the following:]

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BASES (continued)

LCO (continued)

[For this facility, the support systems required OPERABLE to ensure Containment Spray System and Containment Cooling System OPERABILITY are as follows:]

[ ]

[For this facility, those required support systems which upon their failure do not declare the containment spray and cooling systems inoperable and their justification are as follows:]

In addition, each Containment Spray System and Containment Cooling System must satisfy all the performance and physical arrangement requirements set forth by the SRs in order to be considered OPERABLE.

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 MODE 3 or 4, individual plants may justify removal of the Containment Spray System from operation to support Shutdown Cooling System operation. In this condition, the Containment Cooling System must remain OPERABLE. Justification of Containment Spray System removal will be addressed on a plant-specific basis.

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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BASES (continued)

ACTIONS

A.1

If one containment spray train is inoperable, it must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing greater than 100% of the heat-removal needs (for the condition of one containment spray train inoperable) after an accident. The 7-day Completion Time was chosen in light of the redundant heat-removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of DBA occurring during this period.

B.1

If one of the required containment cooling trains is inoperable, it must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing greater than 100% of the heat-removal needs (for the Condition of one containment cooling train inoperable) after an accident. The 7-day Completion Time was chosen based on the same reasons as given in Required Action A.1.

C.1

With two of the required Containment spray trains inoperable, one must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing greater than 100% of the heat-removal needs after an accident. The 72-hour Completion Time was chosen in light of the redundant heat-removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System, reasonable time for repairs, and low probability of DBA occurring during this period.

D.1 and D.2

If one of the required containment spray trains is inoperable and one of the required containment cooling trains is inoperable, the inoperable containment spray train or the inoperable Containment cooling train must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing > 100% of the

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BASES (continued)

ACTIONS (continued)

heat-removal needs (for the Condition of one containment spray train inoperable and one containment cooling train inoperable) after an accident. The 72-hour Completion Time was chosen based on the same reasons as those given in Action C.1.

E.1

If two of the required con nment cooling trains are inoperable, one of the req ed containment cooling trains must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing > 100% of the heat-removal needs after an accident. The 72-hour Completion Time was chosen based on the same reasons as given in Action C.1.

F.1

Any combinations of three or more containment spray and containment cooling trains inoperable, the plant is in a condition outside the accident analysis. Therefore LCO 3.0.3 must be entered immediately.

G.1 and G.2

The plant must be placed in a MODE in which the LCO does not apply if any of the Required Actions and associated Completion Times for Condition A, B, C, D, or E of this LCO are not met. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable time, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.6B.1

Verifying the correct alignment for manual, power-operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

be in the correct positions prior to being secured. The 31-day Frequency of this SR was developed based on Inservice Inspection and Testing Program requirements to perform valve testing at least once per 92 days. This SR does not require testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (only check valves are inside containment) and capable of potentially being mispositioned are in the correct position.

SR 3.6.6B.2

Operating each containment cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and 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 based on the known reliability of the fan units and controls, the two-train redundancy available, and the low probability of significant degradation of the containment cooling train occurring between Surveillances.

SR 3.6.6B.3

Verifying a containment cooling train ESW cooling flow rate of ≥ [700] gpm to each cooling unit provides assurance that the design flow rate assumed in the analyses will be achieved (Ref. 4). The 31-day Frequency of this SR was developed based on Inservice Inspection and Testing Program Requirements to perform testing on safety-related components at least once per 92 days. The Frequency was also 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 3.6.6B.4

Demonstrating that each containment spray pump develops ≥ [ ] psid differential pressure on recirculation ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the American Society of Mechanical Engineers (ASME) Code

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

(Ref. 9). Since the containment spray 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 inspections confirm component OPERABILITY and trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Inspection and Testing Program.

SR 3.6.6B.5 and SR 3.6.6B.6

These SRs require a demonstration that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts on receipt of an actual or simulated containment High-3 pressure signal. The 18-month Frequency was developed considering it is prudent that these Surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unplanned plant transients if the SR is performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The Surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single Surveillance may be used to satisfy both requirements.

SR 3.6.6B.7

This SR ensures that each containment cooling unit actuates on receipt of an actual or simulated safety injection signal. The 18-month Frequency is based on engineering judgment and has been proven acceptable through operating experience. See SR 3.6.6B.5 and SR 3.6.6B.6, above, for further discussion of the basis for the 18-month Frequency.

SR 3.6.6B.8

With the containment spray inlet valves closed and the spray header drained of any solution, low-pressure air or smoke can be blown through test connections. This SR ensures that

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Because of the passive design of the nozzle, a test at the first refueling and then at 10-year intervals is considered adequate to detect degradation in nozzle performance.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."
2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."
4. [Unit Name] FSAR, Sections [ ], "[Accident Analysis]."
5. [Unit Name] FSAR, Section [ ], "[Containment Systems]."
6. [Unit Name] FSAR, Section [ ], "[Title]."
7. [Unit Name] FSAR, Section [ ], "[Title]."
8. [Unit Name] FSAR, Section [ ], "[Title]."
9. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6C Containment Spray System (Ice Condenser)

BASES

BACKGROUND

The Containment Spray System is designed to furnish 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 reduce the release of fission-product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to less than the guidelines of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The Containment Spray System is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," and GDC 40, "Testing of Containment Heat Removal Systems" (Ref. 2), or other documents that were appropriate at the time of licensing (identified on a plant-specific basis).

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the system design bases spray coverage. Each train includes a containment spray pump, one containment spray heat exchanger, spray headers, nozzles, valves, and piping. Each train is powered from a separate ENGINEERED SAFETY FEATURE (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment recirculation sump(s).

The diversion of a portion of the recirculation flow from each train of the Residual Heat Removal (RHR) System to additional redundant spray headers completes the Containment Spray System heat-removal capability. Each RHR train is capable of supplying spray coverage, if required, to supplement the Containment Spray System.

The Containment Spray System and RHR System provide a spray of cold or subcooled borated water into the upper and lower regions of Containment and in dead-ended volumes to limit the containment pressure and temperature during a DBA. The RWST solution temperature is an important factor in

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BASES (continued)

BACKGROUND (continued)

determining the heat-removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the Containment Spray System and RHR heat exchangers. Each train of the Containment Spray System, supplemented by a train of RHR spray, provides adequate spray coverage to meet the system design requirements for containment heat removal.

The Spray Additive System injects a sodium hydroxide (NaOH) solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the Containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically, by a containment High-3 pressure signal, or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two containment spray pumps, and begins the injection phase. 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. The injection phase continues until an RWST level Low-Low alarm is received. The Low-Low alarm for the RWST actuates valves to align the containment spray pump suction to the containment sump and/or signals the operator to manually align the system to recirculation mode. The Containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operation procedures.

The RHR spray operation is initiated manually, when required by the emergency operating procedures, after the Emergency Core Cooling System (ECCS) is operating in the recirculation mode. This additional spray capacity would typically be used after the ice bed has been depleted and in the event that containment pressure rises above a predetermined limit.

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BASES (continued)

BACKGROUND (continued)

The Containment Spray System is an ESF system. It is designed to ensure that the heat-removal capability required during the post-accident period can be attained.

The operation of the Containment Spray System, together with the ice condenser, is adequate to assure pressure suppression during the initial blowdown of steam and water from a DBA. During the post-blowdown period, the Air Return System (ARS) is automatically started. The ARS returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam through the ice condenser, where heat is removed by the remaining ice.

After the ECCS is aligned to the recirculation mode, the RHR spray is aligned to the recirculation mode. The RHR sprays are available to supplement the Containment Spray System, if required, in limiting containment pressure.

The Containment Spray System ensures containment OPERABILITY by limiting the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission-product radioactivity from containment to the environment. Loss of containment integrity could cause site-boundary doses, in the event of a DBA, to exceed values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The Containment Spray System ensures Containment OPERABILITY by limiting the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment OPERABILITY are the loss-of-coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant Containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst-case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the ARS being rendered inoperable (Ref. 2).

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The DBA analyses show that the maximum peak containment pressure of [unit-specific pressure] results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature of [unit-specific temperature] results from the SLB analysis and was calculated to exceed the containment design temperature for [unit-specific time period] during the DBA SLB. The basis of the containment design temperature, however, is to ensure the OPERABILITY of safety-related equipment inside Containment (Ref. 4). Thermal analyses showed that the time interval during which the containment atmosphere temperature exceed the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the DBA SLB.

The modeled Containment Spray System actuation from the Containment OPERABILITY analysis is based on a response time associated with exceeding the containment High-3 pressure signal setpoint to achieving full flow through the containment spray nozzles. A delayed response-time initiation provides conservative analyses of peak calculated Containment temperature and pressure responses. The Containment Spray System total response time of [45] seconds is composed of signal delay, diesel generator startup, and system startup time.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 5).

Inadvertent actuation of the Containment Spray System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure resulted in a containment external pressure load of [unit-specific pressure], which is below the containment design external pressure load.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The Containment Spray System satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

During a DBA, one train of Containment Spray System is required to provide the heat-removal capability assumed in the safety analyses for containment OPERABILITY. Additionally, a minimum of one train of the Containment Spray System, with spray pH adjusted by the Spray Additive System, is required to scavenge iodine fission products from the containment atmosphere and ensure their retention in the containment sump water. To ensure that these requirements are met, two containment spray trains must be OPERABLE with power from two safety-related independent power supplies. Therefore, in the event of an accident, at least one train in each system occurs.

Each Containment Spray System typically includes a spray pump, headers, valves, heat enhancers, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and automatically transferring suction to the containment sump.

[For this facility, an OPERABLE Containment Spray System constitutes the following:]

[For this facility, the support systems required OPERABLE to ensure Containment Spray System OPERABILITY are as follows:]

[For this facility, those required support systems which upon their failure do not require declaring the Containment Spray System inoperable and their justification are as follows:]

In addition, each Containment Spray System must satisfy all the performance and physical arrangement SRs in order to be consider OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in

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BASES (continued)

APPLICABILITY (continued)

containment pressure and temperature requiring the operation of the Containment Spray System.

In MODE 3 or 4, individual plants may justify removal of the Containment Spray System from operation to support Shutdown Cooling System operation. In this condition, the Containment Cooling System must remain OPERABLE. Justification of Containment Spray System removal will be addressed on a plant-specific basis.

In MODES 5 and 6, the probability and consequences of these events are reduced because of the pressure and temperature limitations of these MODES. Thus, the Containment Spray System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

With one Containment Spray System train inoperable, the affected train must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat-removal and iodine-removal needs after an accident. The 72-hour Completion Time was developed taking into account the redundant heat-removal and iodine-removal capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

With two Containment Spray System trains inoperable, the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

B.1 and B.2

In the event the affected containment spray train is not restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 84 hours. The 6 hours allotted to reach MODE 3 is a reasonable time, based on operating experience, to shut down the plant from full power in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5

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BASES (continued)

ACTIONS (continued)

allows 48 hours for restoration of the containment spray train in MODE 3, and 36 hours to reach MODE 5. This is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6C.1

Verifying the correct alignment of manual, power-operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. The 31-day Frequency of this SR was developed based on Inservice Inspection and Testing Program requirements to perform valve testing at least once per 92 days. 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 3.6.6C.2

Demonstrating that each containment spray pump develops ≥ [ ] psid differential pressure on recirculation ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the American Society of Mechanical Engineers (ASME) Code (Ref. 6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Inspection and Testing Program.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6C.3 and SR 3.6.6C.4

These SRs ensure that each automatic Containment spray valve actuates to its correct position and each containment spray pump starts on receipt of an actual or simulated containment spray actuation signal. The 18-month Frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The Surveillance of containment sump isolation valves is also required by SR 3.6.6.3. A single Surveillance may be used to satisfy both requirements.

SR 3.6.6C.5

With the containment spray inlet valves closed and the spray header drained of any solution, low-pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Because of the passive design of the nozzle, a 10-year test interval is considered adequate to detect degradation in nozzle performance.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100, "Determination of Exclusion Area Low Population Zone and Population Center Distance."

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Criteria for Nuclear Power Plants."

3. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

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BASES (continued)

REFERENCES (continued)

4. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

5. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

6. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6D Quench Spray (QS) System (Subatmospheric)

BASES

BACKGROUND

The QS System is designed to furnish containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. The QS System, operating in conjunction with the Recirculation Spray (RS) System, is designed to cool and depressurize the containment structure to subatmospheric pressure in less than 60 minutes following a Design Basis Accident (DBA). Reduction of containment pressure and the iodine-removal capability of the spray limit the release of fission-product radioactivity from containment to the environment, in the event of a DBA, to within 10 CFR 100 limits (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

The QS System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a spray pump, spray headers, nozzles, valves, and piping and is powered from a separate ENGINEERED SAFETY FEATURE (ESF) bus. The Refueling Water Storage Tank (RWST) supplies borated water to the QS System.

The QS System is actuated either automatically by a containment High-High pressure signal, or manually. The QS System provides a spray of cold borated water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. Each train of the QS System provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission-product removal. The QS System also provides flow to the containment sump to improve the net positive suction head available to the RS System pumps.

The Spray Additive System injects a sodium hydroxide (NaOH) solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The NaOH added to the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of

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BASES (continued)

BACKGROUND (continued)

iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The QS System is a containment ESF System. It is designed to ensure that the heat-removal capability required during the post-accident period can be attained. Operation of the QS System and RS System provide the required heat-removal capability to limit post-accident conditions to less than the containment design values and depressurize the containment structure to subatmospheric pressure in less than 60 minutes following a DBA.

The QS System ensures containment OPERABILITY by limiting the temperature and pressure that could be expected following a DBA. Ensuring containment OPERABILITY limits leakage of fission-product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause site-boundary doses, in the event of a DBA, to exceed values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The QS System ensures containment OPERABILITY by limiting the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment OPERABILITY are the loss-of-coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, with respect to containment ESF Systems, assuming the loss of one ESF bus, which is the worst-case single active failure, resulting in one train of the QS System and RS System inoperable.

During normal operation, the containment internal pressure is varied to maintain the capability to depressurize the containment to a subatmospheric pressure in less than 60 minutes after a DBA. This capability and the variation of containment pressure are functions of the service water temperature, the RWST water temperature, and the containment air temperature.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The DBA analyses (Ref. 2) show that the maximum peak containment pressure of [unit-specific pressure] results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature of [unit-specific temperature] results from the SLB analysis and was calculated to exceed the containment design temperature for [unit-specific time period] during the transient. The basis of the containment design temperature, however, is to ensure OPERABILITY of safety-related equipment inside containment (Ref. 3). Thermal analyses showed that the time interval during which the containment atmosphere temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB.

The modeled QS System actuation from the containment OPERABILITY analysis is based upon a response time associated with exceeding the containment High-High pressure signal setpoint to achieving full flow through the spray nozzles. A delayed-response-time initiation provides conservative analyses peak calculated containment temperature and pressure responses. The QS System total response time of [66] seconds comprises the signal delay, diesel generator startup time, and system startup time.

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 System (ECCS) during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 4).

Inadvertent actuation of the QS System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure resulted in a containment external pressure load of [unit-specific pressure], which is below the containment design external-pressure load.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The QS System satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

During a DBA, one train of the QS System is required to provide the heat-removal capability assumed in the safety analyses for containment. In addition, one QS System train, with spray pH adjusted by the Spray Additive System, is required to scavenge iodine fission products from the containment atmosphere and ensure their retention in the containment sump water. To ensure that these requirements are met, two QS System trains must be OPERABLE with power from two safety-related, independent power supplies. Therefore, in the event of an accident, at least one train in each system will operate, assuming that the worst-case single active failure occurs.

Each QS System includes a spray pump, headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST.

[For this facility, an OPERABLE QS System constitutes of the following:]

[For this facility, the support systems required OPERABLE to ensure QS System OPERABILITY are as follows:]

[For this facility, those required support systems which upon their failure do not require declaring the QS System inoperable and their justification are as follows:]

In addition, each QS System must satisfy all the performance and physical arrangement requirements set forth by the SRs in order to be considered OPERABLE.

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 QS System.

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(continued)

BASES (continued)

APPLICABILITY (continued)

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 QS System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

If one QS System train is inoperable, it must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat-removal and iodine-removal needs after an accident. The 72-hour Completion Time was developed taking into account the redundant heat-removal and iodine-removal capabilities afforded by the OPERABLE train, and the low probability of DBA occurring during this period.

With two QS train inoperables, the plant is in a Condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the Required Actions and associated Completion Times are not met, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.6D.1

Verifying the correct alignment of manual, power-operated, and automatic valves, excluding check valves, in the QS System provides assurance that the proper flow path exists for QS System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to being secured. The 31-day Frequency of this SR was developed based upon Inservice Inspection and Testing Program requirements to perform valve testing at least once

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

per 92 days. 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 3.6.6D.2

Demonstrating that each QS pump develops ≥ [ ] psid differential pressure on recirculation ensures that QS pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the American Society of Mechanical Engineers (ASME) Code (Ref. 5). Since the QS System pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Inspection and Testing Program.

SR 3.6.6D.3 and SR 3.6.6D.4

These SRs ensure that each automatic containment spray valve actuates to its correct position and each containment spray pump starts on receipt of an actual or simulated containment spray actuation signal. The 18-month Frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. This is because of the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed on an 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6D.5

With the containment spray inlet valves closed and the spray header drained of any solution, low-pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Due to

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

the passive nature of the design of the nozzle, a 10-year test interval is considered adequate to detect degradation in the performance of the nozzles.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

3. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

4. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6E Recirculation Spray (RS) System (Subatmospheric)

BASES

BACKGROUND

The RS System, operating in conjunction with the Quench Spray (QS) System, is designed to limit the post-accident pressure and temperature in the containment to less than the design values, and to depressurize the containment structure to a subatmospheric pressure in less than 60 minutes following a Design Basis Accident (DBA). The reduction of containment pressure and the removal of iodine from the containment atmosphere by the spray limit the release of fission-product radioactivity from containment to the environment, in the event of a DBA, to within 10 CFR 100 limits (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

The RS System consists of two separate trains of equal capacity, each capable of meeting the design and accident analysis bases. Each train includes one RS subsystem outside containment and one RS subsystem inside containment. Each subsystem consists of one 50%-capacity spray pump, one spray cooler, one 180°-coverage spray header, nozzles, valves, piping, instrumentation, and controls. Each outside RS subsystem also includes a casing cooling pump with its own valves, piping, instrumentation, and controls. The two outside RS subsystems' spray pumps are located outside containment and the two inside RS subsystems' spray pumps are located inside containment. Each RS train (one inside and one outside RS subsystem) is powered from a separate ENGINEERED SAFETY FEATURE (ESF) bus. Each train of the RS System provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission-product removal.

The two casing cooling pumps and common casing cooling tank are designed to increase the net positive suction head (NPSH) available to the outside RS pumps by injecting cold water into the suction of the spray pumps. The casing cooling water tank contains 116,500 gal of chilled and borated water. Each casing cooling pump supplies one outside spray pump with cold borated water from the casing cooling water tank. The casing cooling pumps are considered part of the outside RS subsystems. Each casing cooling pump is powered from a separate ESF bus.

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BASES (continued)

BACKGROUND (continued)

The RS provides a spray of subcooled water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. Upon receipt of a High-High containment pressure signal, the two casing cooling pumps start and the RS pump suction and discharge valves receive an open signal to assure the valves are open. After a [195]-second time delay, the inside RS pumps start, and after a [210]-second time delay, the outside RS pumps start. The RS pumps take suction from the containment sump and discharge through their respective spray coolers to the spray headers and into the containment atmosphere. Heat is transferred from the containment sump water to service water in the spray coolers.

The Spray Additive System injects a sodium hydroxide (NaOH) solution into the suction of the QS System pumps. The NaOH added to the QS System spray ensures an alkaline pH for the solution recirculated in the containment sump. The resulting alkaline pH of the RS spray (pumped from the sump) enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The RS is a containment ESF System. It is designed to ensure that the heat-removal capability required during the post-accident period can be attained. Operation of the QS and RS systems provides the required heat-removal capability to limit post-accident conditions to less than the containment design values and depressurize the containment structure to subatmospheric pressure in less than 60 minutes following a DBA.

The RS ensures containment OPERABILITY by limiting the temperature and pressure that could be expected following a DBA. Ensuring containment OPERABILITY limits leakage of fission-product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause site-boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The RS System ensures containment OPERABILITY by limiting the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment OPERABILITY are the loss-of-coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients; DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed assuming the loss of one ESF bus, which is the worst-case single active failure, resulting in one train of QS and RS Systems being rendered inoperable (Ref. 2).

The peak containment pressure following a high-energy line break is affected by the initial total pressure and temperature of the containment atmosphere and the QS System operation. Maximizing the initial containment total pressure and average atmospheric temperature maximizes the calculated peak pressure. The heat-removal effectiveness of the QS System spray is dependent on the temperature of the water in the refueling water storage tank (RWST). The time required to depressurize the containment and the capability to maintain it depressurized below atmospheric pressure depend on the functional performance of the QS and RS Systems and the service water temperature. When the Service Water temperature is elevated, it is more difficult to depressurize the containment in 60 minutes since the heat-removal effectiveness of the RS System is limited.

During normal operation, the containment internal pressure is varied to maintain the capability to depressurize the containment to a subatmospheric pressure in less than 60 minutes after a DBA. This capability and the variation of containment pressure are functions of service water temperature, RWST water temperature, and the containment air temperature.

The DBA analyses show that the maximum peak containment pressure of [ ] psig results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum [unit-specific value] peak containment atmosphere temperature results from the SLB analysis and is calculated to exceed the containment design temperature for [unit-specific time period] during the transient. The basis of the containment design temperature, however, is to ensure

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(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

OPERABILITY of safety-related equipment inside containment (Ref. 3).

Thermal analyses show that the time interval during which the containment atmosphere temperature exceeds the containment design temperature is short enough that equipment surface temperatures remain below the design temperature. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB.

The RS System actuation model from the containment OPERABILITY analysis is based upon a response time associated with exceeding the High-High containment pressure signal setpoint to achieving full flow through the RS System spray nozzles. A delay in response-time initiation provides conservative analyses of peak calculated containment temperature and pressure. The RS System's total response time of 300 seconds comprises the signal delay, diesel generator startup time, and system startup time.

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 System (ECCS) during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 4).

The RS System satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

During a DBA, one train (two subsystems) of the RS System is required to provide the minimum heat-removal capability assumed in the safety analysis for containment OPERABILITY. To ensure that this requirement is met, four RS subsystems must be OPERABLE. This will ensure that at least one train will operate assuming the worst-case single failure occurs, which is in the ESF power supply.

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BASES (continued)

LCO (continued)

[For this facility, an OPERABLE RS subsystem consists of the following:]

[For this facility, the following support systems are required OPERABLE to ensure RS System OPERABILITY.]

[For this facility, those required support systems which upon their failure do not require declaring the RS System inoperable and their justification are as follows:]

[For this facility, an OPERABLE casing cooling tank consists of the following:]

[For this facility, the following support systems are required OPERABLE to ensure casing cooling tank OPERABILITY:]

[For this facility, those required support systems which upon their failure do not require declaring the casing cooling tank inoperable and their justification are as follows:]

In addition, each RS subsystem and casing cooling tank must satisfy all the physical and performance requirements set forth by the SRs in order to be considered OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the RS System.

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 RS System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

With one of the RS subsystems inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing greater than 100% of the heat-removal needs (i.e.,

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

150% when one RS subsystem is inoperable) after an accident. The 7-day Completion Time was developed taking into account the redundant heat-removal capabilities afforded by combinations of the RS and QS systems and the low probability of a DBA occurring during this period.

B.1

With two of the required RS subsystems inoperable in the same train, at least one of the inoperable RS subsystems must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat-removal needs after an accident. The 72-hour Completion Time was developed taking into account the redundant heat-removal capability afforded by the OPERABLE subsystems, reasonable time for repairs, and the low probability of DBA occurring during this period.

C.1

With two of the inside RS subsystems inoperable, at least one of the inoperable subsystems must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat-removal needs after an accident. The 72-hour Completion Time was chosen based on the same reasons as given in Action B.1.

D.1

With two of the outside RS subsystems inoperable, at least one of the inoperable subsystems must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat-removal needs after an accident. The 72-hour Completion Time was chosen based on the same reasons as given in Action B.1.

E.1

With the casing cooling tank inoperable, the NPSH available to the outside RS subsystem pumps may not be sufficient. The inoperable casing cooling tank must be restored to OPERABLE status within 72 hours. The components in this

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ACTIONS (continued)

degraded condition are capable of providing 100% of the heat-removal needs after an accident. The 72-hour Completion Time was chosen based on the same reasons as given in Action B.1.

F.1 and F.2

If the inoperable RS subsystem(s) or the casing cooling tank cannot be restored in the required Completion Time, the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 84 hours. The 6 hours allotted for reaching MODE 3 is a reasonable amount of time, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the RS subsystem or casing cooling tank in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

G.1

With three RS subsystems inoperable, the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6E.1

Verifying that the casing cooling tank solution temperature is within the specified tolerances provides assurance that the water injected into the suction of the outside RS pumps will increase the NPSH available as per design. The 24-hour Frequency of this SR was developed considering operating experience related to the parameter variations and instrument drift during the applicable MODES. 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 condition.

The required number of OPERABLE channels is established in LCO [ ] or SR [ ].

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6E.2

Verifying the casing cooling tank contained water volume provides assurance that sufficient water is available to support the outside RS subsystem pumps during the time they are required to operate. The 7-day Frequency of this SR was developed considering operating experience related to the parameter variations and instrument drift during the applicable MODES. Furthermore, the 7-day Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal condition.

SR 3.6.6E.3

Verifying the boron concentration of the solution in the casing cooling tank provides assurance that borated water added from the casing cooling tank to RS subsystems will not dilute the solution being recirculated in the containment sump. The 7-day Frequency of this SR was developed considering the known stability of stored borated water and the low probability of any source of diluting pure water.

SR 3.6.6E.4

Verifying the correct alignment of manual, power-operated, and automatic valves, excluding check valves, in the RS System and casing cooling systems provides assurance that the proper flow path exists for operation of the RS System. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified as being in the correct position prior to being secured. The 31-day Frequency of this SR was based on Inservice Inspection and Testing Program requirements for performing valve testing at least once per 92 days. 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 3.6.6E.5

Demonstrating that each RS pump develops ≥ [ ] psid and casing cooling pump ≥ [ ] psid differential pressure on recirculation ensures that these pumps' performance has not

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SURVEILLANCE REQUIREMENTS (continued)

degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the American Society of Mechanical Engineers (ASME) Code (Ref. 5). Since the QS System pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve, and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Inspection and Testing Program.

SR 3.6.6E.6

These SRs ensure that each automatic valve actuates and that the RS System and casing cooling pumps start upon receipt of an actual or simulated High-High containment pressure signal. Start delay times are also verified for the RS System pumps. The 18-month Frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed on the 18-month Frequency. Therefore, the Frequency is considered to be acceptable from a reliability standpoint.

SR 3.6.6E.7

This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment will meet its design bases objective. An air or smoke test is performed through each spray header. Due to the passive design of the spray header and its normally dry state, a Frequency of 10 years is considered adequate for detecting degradation in the performance of the nozzles.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

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BASES (continued)

REFERENCES (continued)

2. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

3. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

4. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant components," American Society of Mechanical Engineers, New York.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

BASES

BACKGROUND

The Spray Additive System is a subsystem of the containment Spray System, which assists in reducing the iodine fission-product inventory in the containment atmosphere. Reduction of the iodine fission-product inventory limits the site-boundary exposure resulting from a Design Basis Accident (DBA) to within the thyroid-dose guidelines of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

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. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the preferred spray additive. The NaOH added to the spray also ensures a pH value of between 8.5 and 11.0 of the solution recirculated from the containment sump. This pH band minimizes the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

Eductor Feed Systems Only

The Spray Additive System consists of one spray additive tank that is shared by the two trains of spray additive equipment. Each train of equipment provides a flow path from the spray additive tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line. The eductors are designed to ensure that the pH of the spray mixture is between 8.5 and 11.0.

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BASES (continued)

BACKGROUND (continued)

Gravity Feed Systems Only

The Spray Additive System consists of one spray-additive tank, two parallel redundant motor-operated valves in the line between the additive tank and the Refueling Water Storage Tank (RWST), instrumentation, and recirculation pumps. The NaOH solution is added to the spray water by a balanced gravity feed from the additive tank through the connecting piping into a weir within the RWST. There, it mixes with the borated water flowing to the spray pump suction. Because of the hydrostatic balance between the two tanks, the flow rate of the NaOH is controlled by the volume per foot of height ratio of the two tanks. This ensures a spray mixture pH that is ≥ 8.5 and ≤ 11.0.

The containment spray system actuation signal opens the valves from the spray-additive tank to the spray pump suctions or the containment spray pump start signal opens the valves from the spray chemical addition tank (SCAT) after a 5-minute delay. The 28% to 31% NaOH solution is drawn into the spray pump suctions. The SCAT capacity provides for the addition of NaOH solution to all of the water sprayed from the RWST into containment. The percent solution and volume of solution sprayed into containment ensures a long-term containment sump pH of ≥ 9.0 and ≤ 9.5. This ensures the continued iodine-retention effectiveness of the sump water during the recirculation phase of spray operation and also minimizes the occurrence of chloride-induced stress corrosion cracking of the stainless-steel recirculation piping.

The Spray Additive System reduces the iodine fission-product inventory in the containment atmosphere. Loss of the Spray Additive System could cause site-boundary radiation exposures resulting from a DBA to exceed the dose guidelines in the licensing basis.

APPLICABLE SAFETY ANALYSES

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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value of [0.1] air weight percent per day following the accident. The analysis assumes that 100% of containment 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 B 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 SCAT volume is added to the remaining Containment Spray System flow path.

The potential radiological consequences of the DBA have been analyzed for the 2-hour dose at the exclusion-area boundary and for the duration of the accident at the low-population-zone outer boundary. The resultant doses are within the guideline values of the licensing basis.

The Spray Additive System satisfies Criterion 3 of the NRC Interim Policy Statement.

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 RWST to the containment sump and to raise the average spray solution pH to a level conducive to iodine removal, namely, to between [7.2 and 11.0]. This pH range maximizes the effectiveness of the iodine-removal mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

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BASES (continued)

LCO (continued)

[For this facility, an OPERABLE Spray Additive System constitutes the following:]

[ ]

[For this facility, the following support systems are required OPERABLE to ensure Spray Additive System OPERABILITY:]

[For this facility, those required support systems which upon their failure do not require declaring the Spray Additive system inoperable and their justification are as follows:]

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the Spray Additive System. The OPERABILITY of the Spray Additive System is essential to limit post-accident release of radioactive material to the environment to within the limits in the licensing basis.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitation in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine-removal enhancement are 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.

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BASES (continued)

ACTIONS (continued)

B.1 and B.2

If the Spray Additive System is not restored to OPERABLE status within the associated Completion Time the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 84 hours. The 6 hours allotted to reach MODE 3 is a reasonable time, based on operating experience, to shut down the plant in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System 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. The 31-day Frequency of this SR was developed based upon Inservice Inspection and Testing Program requirements to perform valve testing at least once per 92 days. 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 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the SCAT 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Additive System. The 184-day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal plant operations). Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the SCAT and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184-day Frequency is sufficient to ensure that the concentration level of NaOH in the SCAT 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 demonstrates that each automatic valve in the Spray Additive System flow path actuates to its correct position. The 18-month Frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed on 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 demonstrated once every 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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BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."
2. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Monitors—MODES 1 & 2 (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

BASES

BACKGROUND

Hydrogen monitors are required to monitor the hydrogen concentration in the containment following a loss-of-coolant accident (LOCA) or steam line break (SLB) in containment. Hydrogen may accumulate or pocket within containment following a LOCA as a result of a metal-steam reaction involving the zirconium fuel cladding and the reactor coolant, radiolytic decomposition of the post-accident emergency cooling solutions, corrosion of metals by solutions used for emergency cooling and containment spray, and hydrogen in the Reactor Coolant System (RCS) at the time of the LOCA. The primary source of hydrogen production after a SLB is corrosion of aluminum by caustic solutions (containment spray). The lower flammability limit of hydrogen is 4.1 volume percent (v/o) (Ref. 1). Should the lower flammability limit be exceeded, hydrogen ignition could occur. This could lead to overpressurization of containment, resulting in a breach of containment OPERABILITY, unacceptably high containment leakage and offsite doses, and damage to safety-related equipment inside containment. To ensure that this limit is not exceeded, a conservative control limit for hydrogen inside containment has been set at 3.5 v/o. When the 3.5 v/o control limit is reached, some means of hydrogen control, either purging or operation of the hydrogen recombiners, must be initiated to reduce hydrogen concentration in the containment.

The hydrogen monitors are a post-accident Type C, Category 1, instrument. As such they will function to allow monitoring of hydrogen following a LOCA or SLB in containment.

Two independent hydrogen monitors have been provided; each is powered from a separate vital AC power source. The monitors are manually actuated from their control panels and are required to operate after a LOCA. The monitors, when actuated, will continuously monitor hydrogen concentration levels between 0% and 10%. Both monitors have the capability to interface with two areas that have been

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BASES (continued)

BACKGROUND (continued)

selected to provide a representative sample of the containment atmosphere following an accident.

The hydrogen monitors provide the capability to measure the hydrogen concentration in containment so that required operator actions (e.g., actuate the hydrogen recombiners or Hydrogen Purge System in accordance with emergency procedures) may be taken to prevent the hydrogen concentration from exceeding the flammability limit of 4.1 v/o. Accurate measurement of hydrogen is attained at containment pressures up to 50 psi and temperatures to 445°F (Ref. 2). The information provided by these monitors is used by the plant operators to determine when Hydrogen Purge System or hydrogen recombiner actuation is required to maintain the hydrogen concentration below the lower flammability limit. This will eliminate the potential for a breach of containment due to a hydrogen-oxygen reaction.

APPLICABLE SAFETY ANALYSES

The hydrogen monitors monitor the post-accident containment atmosphere and provide an indication of containment hydrogen concentration in the post-accident containment atmosphere. This information is utilized by the operators to determine when the combustible gas control systems (Hydrogen Purge System and hydrogen recombiners) should be actuated, if needed, to maintain the hydrogen concentration below the flammability limit.

Assumptions recommended in Reference 1 are used to maximize the amount of hydrogen calculated. The calculations confirm that when mitigating systems are actuated in accordance with the emergency operating procedures, the peak hydrogen concentration in containment is less than 4.1 v/o.

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

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

b. Radiolytic decomposition of water in the RCS and the containment sump;

(continued)

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System (ECCS) solutions.

Hydrogen monitors satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

Two hydrogen monitors must be OPERABLE with power from two independent safety-related power supplies. Either analyzer is capable of obtaining and analyzing a representative sample from the containment dome and the emergency reactor building recirculation ventilation duct. This assures operation of at least one hydrogen monitor in the event of a worst-case single active failure. Operation of at least one hydrogen monitor will provide the operator with information to enable action to be taken to prevent the containment post-LOCA hydrogen concentration from exceeding the flammability limit.

[For this facility, the following support systems are required to be OPERABLE to ensure hydrogen monitor OPERABILITY:]

[For this facility, those required support systems which upon their failure do not require declaring the hydrogen monitor inoperable and their justification are as follows:]

[For this facility, an OPERABLE hydrogen monitor constitutes the following:]

APPLICABILITY

In MODES 1 and 2, two hydrogen monitors provide the operator with the capability to measure hydrogen concentration in containment assuming a worst-case single active failure and, if required, allow action to be taken to control the hydrogen concentration within containment below its flammability limit of 4.1% following a LOCA (Ref. 2). This ensures containment OPERABILITY and prevents damage to

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(continued)

BASES (continued)

APPLICABILITY (continued)

safety-related equipment and instrumentation located within containment.

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

In MODES 5 and 6, the probability and consequences of a LOCA are reduced because of the pressure and temperature limitations of these MODES. Therefore, the hydrogen monitors are not required in these MODES to ensure containment OPERABILITY.

ACTIONS

A.1

With one hydrogen monitor inoperable, the inoperable analyzer must be restored to OPERABLE status within 30 days. The 30-day Completion Time is based on the low probability of failure of the other redundant hydrogen monitor, the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit, the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit, and the availability of the hydrogen recombiners, the Hydrogen Purge System, and the Post-Accident Sampling System.

Concurrent failure of two hydrogen monitors within a 30-day period is considered to be a low-probability event. If such double failures were to occur, it would be indicative of poor hydrogen monitor reliability and would result in the loss of functional capability. Therefore, LCO 3.0.3 must be entered immediately.

B.1

If an inoperable hydrogen monitor cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 in

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BASES (continued)

ACTIONS (continued)

6 hours. The 6 hours allotted to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

AN ANALOG CHANNEL OPERATIONAL TEST is performed on each hydrogen monitor every 92 days to ensure the entire channel will perform its intended function. The 92-day Frequency is based on the reliability of the hydrogen monitors, which has been demonstrated to be acceptable through operating experience.

[For this facility, an ANALOG CHANNEL OPERATIONAL TEST constitutes the following:]

SR 3.6.8.2

Performance of a CHANNEL CALIBRATION on the hydrogen monitors using sample gases ensures the OPERABILITY of the analyzers is maintained. A typical CHANNEL CALIBRATION includes a minimum of two data points to verify accuracy of the analyzers over the range of interest. The sample gases used for performing the Surveillances are nominally 1 v/o hydrogen, $\geq$ 0.98 and $\leq$ 1.02 (balance nitrogen), and nominally 4 v/o hydrogen, $\geq$ 3.92 and $\leq$ 4.08 (balance nitrogen). The lower hydrogen flammability limit has been assumed as 4.1 v/o hydrogen in air or steam-air atmospheres (Ref. 1). Therefore, calibration with these sample gases helps ensure accurate information regarding containment hydrogen concentrations up to and including the flammability limit is available to the operators following a LOCA. [For this facility, the 18-month Frequency has been shown to be acceptable through operating experience and is further justified because of other checks performed during the CHANNEL FUNCTIONAL TEST, which convey that proper calibration of hydrogen monitors is being maintained.]

REFERENCES

1. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978.

2. [Unit Name] FSAR, Appendix [ ], "[Title]."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Hydrogen Recombiners—MODES 1 & 2 (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

BASES

BACKGROUND

The hydrogen recombiners support containment OPERABILITY in post-accident environments by eliminating the potential breach of containment due to a hydrogen-oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), 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 remains in containment, thus eliminating any discharge to the environment.

Two independent hydrogen recombiner systems are provided. Each consists of controls located in the control room, a power supply and a recombiner. 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 unit. 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 FEATURE bus, and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners ensure containment OPERABILITY by providing the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 v/o following a DBA. This control would prevent a containment-wide hydrogen burn, thus ensuring containment OPERABILITY and minimizing challenges to the OPERABILITY of safety-related equipment located in containment. The limiting DBA relative to hydrogen generation is a LOCA.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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 and Emergency Core Cooling System 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 3 are used to maximize the amount of hydrogen calculated. As such, the hydrogen recombiners are designed to control an amount of hydrogen generation in containment considerably in excess of the amount that would be calculated from the limiting DBA LOCA.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 3). The Hydrogen Purge System is similarly designed such that one of two redundant trains is an adequate backup to the redundant hydrogen recombiners.

The hydrogen recombiners satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

Two hydrogen recombiners must be OPERABLE with power from two independent safety-related power supplies. Each typically consists of controls, power supply and recombiner.

[For this facility, an OPERABLE hydrogen recombiner consists of the following:]

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BASES (continued)

LCO (continued)

Operation with at least one hydrogen recombiner ensures that the post-LOCA hydrogen concentration can be prevented from exceeding the flammability limit. Unavailability of both hydrogen recombiners might lead to hydrogen generation in an amount sufficient (the flammability limit exceeded) to react with oxygen following the accident. The reaction could take place fast enough to lead to high temperatures and overpressurization of containment and, as a result, breach containment or cause containment leakage rates above those assumed in the safety analyses. Damage to safety-related equipment located in containment could also occur.

[For this facility, the following support systems are required OPERABLE to ensure hydrogen recombiner OPERABILITY:]

[For this facility, those required support systems which upon their failure do not require declaring the hydrogen recombiner inoperable and their justification are as follows:]

APPLICABILITY

Requiring OPERABILITY in MODES 1 and 2 for the hydrogen recombiners ensures their immediate availability after the safety injection and scram actuated on a LOCA or SLB initiation. In the post-accident LOCA or SLB environment, one hydrogen recombiner is 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. This ensures containment OPERABILITY and prevents damage to safety-related equipment and instruments located within containment.

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

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

(continued)

BASES (continued)

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. The 30-day Completion Time is based on the low probability of the occurrence of a LOCA or SLB that would generate hydrogen in amounts capable of exceeding the flammability limit, the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit, and the low probability of failure of the OPERABLE hydrogen recombiner.

Concurrent failure of two hydrogen recombiners within a 30-day period is considered to be a low-probability event. If such a double failure were to, it would be indicative of poor hydrogen recombiner reliability and would result in the loss of functional capability. Therefore, LCO 3.0.3 must be entered immediately.

B.1

The plant must be placed in a MODE in which the LCO does not apply if the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 within 6 hours. The 6 hours allotted to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to $\geq$ 700°F in $\leq$ 90 minutes. After reaching 700°F, the power is increased to maximum power for approximately 2 minutes and power is verified to be $\geq$ 60 kW.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The 18-month Frequency for this SR was developed considering such factors as the following:

a. The incidence of hydrogen recombiners failing the SR in the past is low;

b. Even when hydrogen recombiner failure has been detected, there has been, in all instances, a backup available either from the other recombiner or from a diverse system [Hydrogen Purge System]; and

c. Since the hydrogen recombiner is manually started many hours after a LOCA occurs, there is time available to either restore a recombiner to OPERABLE status, or activate an alternative.

SR 3.6.9.2

This SR ensures there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failure involves loss of power, blockage of the internal flow, 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 such factors as the following:

a. The incidence of hydrogen recombiners failing the SR in the past is low;

b. Even when hydrogen recombiner failure has been detected, there has been, in all instances, a backup available either from the other recombiner or from a diverse system [Hydrogen Purge System]; and

c. Since the hydrogen recombiner is manually started many hours after a LOCA occurs, there is time available to either restore a combiner to OPERABLE status or activate an alternative.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.3

This SR requires performance of a resistance-to-ground test of 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 such factors as the following:

a. The incidence of hydrogen recombiners failing the SR in the past is low;

b. Even when hydrogen recombiner failure has been detected, there has been, in all instances, a backup available either from the other recombiner or from a diverse system [Hydrogen Purge System]; and

c. Since the hydrogen recombiner is manually started many hours after a LOCA occurs, there is time available to either restore a recombiner to OPERABLE status or activate an alternative.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.44, "Standards for Combustible Gas Control Systems in Light Water-Cooled Power Reactors."

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

3. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Hydrogen Mixing System (HMS)—MODES 1 & 2 (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

BASES

BACKGROUND

The HMS supports containment OPERABILITY in post-accident environments by reducing the potential for breach of containment due to a hydrogen-oxygen reaction. The HMS ensures containment OPERABILITY by providing a uniformly mixed post-accident containment atmosphere, thereby minimizing the potential for local hydrogen burns due to a pocket of hydrogen above the flammable concentration. Maintaining a uniformly mixed containment atmosphere also ensures that the hydrogen monitors will give an accurate measure of the bulk hydrogen concentration and give the operator the capability of preventing the occurrence of a bulk hydrogen burn inside containment. Containment OPERABILITY limits leakage of fission-product radioactivity from containment to the environment.

The post-accident HMS is an ENGINEERED SAFETY FEATURE (ESF) and is designed to withstand a loss-of-coolant accident (LOCA) without loss of function. The system has two independent trains, each consisting of two fans with their own motors and controls. Each train is sized for [4000] cfm. The two trains are initiated automatically on a Phase A containment isolation signal. The automatic action is to start the non-operating hydrogen mixing fans on slow speed, and shift the operating hydrogen mixing fans (if any) to slow speed. Each train is powered from a separate emergency power supply. Since each train fan can provide 100% of the mixing requirements, the system will provide its design function with a limiting single active failure.

Air is drawn from the steam generator compartments by the locally mounted mixing fans and is discharged toward the upper regions of the containment. This complements the air patterns established by the containment air coolers, which take suction from the operating floor level and discharge to the lower regions of the containment, and the containment spray, which cools the air and causes it to drop to lower elevations. The systems work together such that potentially stagnant areas where hydrogen pockets could develop are eliminated.

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BASES (continued)

BACKGROUND (continued)

When performing their post-accident hydrogen mixing function, the hydrogen mixing fans operate on slow speed to prevent motor overload in a post-accident high pressure environment. The design flow rate on slow speed is based on the minimum air distribution requirements to eliminate stagnant hydrogen pockets. Each train is redundant (full capacity) and is powered from an independent ESF bus. The hydrogen mixing fans may be operated on fast speed during normal operation when a containment air cooler is taken out of service. As such, the design flow rate of the hydrogen mixing fans for high-speed operation is based on air distribution requirements during such normal operation.

APPLICABLE SAFETY ANALYSES

The HMS ensures containment OPERABILITY by providing the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 volume percent (v/o) (Ref. 1) following a DBA. This control would prevent a containment-wide hydrogen burn, thus ensuring containment OPERABILITY and minimizing challenges to the OPERABILITY of safety-related equipment located in containment. The limiting DBA relative to hydrogen generation is a LOCA.

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

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

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

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 and Emergency Core Cooling System 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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Reference 1 are used to maximize the amount of hydrogen calculated. As such, the HMS is designed to control an amount of hydrogen generation in containment in excess of the amount that would be calculated from the limiting DBA LOCA (Ref. 2).

The HMS satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

Two HMS trains must be OPERABLE, with power from two independent, safety-related power supplies. Each train typically consists of two fans with their own motors and controls and is automatically initiated by a Phase A containment isolation signal.

[For this facility, an OPERABLE HMS train constitutes the following:]

Operation with at least one HMS train provides the capability of controlling the bulk hydrogen concentration in containment without exceeding the flammability limit. Unavailability of both HMS trains might lead to containment-wide hydrogen burns.

[For this facility, the following support systems are required OPERABLE to ensure HMS OPERABILITY:]

[For this facility, those required support systems which upon their failure do not require declaring the HMS inoperable and their justification are as follows:]

APPLICABILITY

Requiring OPERABILITY in MODES 1 and 2 for the HMS ensures its immediate availability after the safety injection and scram actuated on a LOCA or SLB initiation. In the post-accident LOCA or SLB environment, the two HMS trains ensure the capability to prevent localized hydrogen concentrations above the flammability limit of 4.1 v/o in containment assuming a worst-case single active failure. This ensures containment OPERABILITY and prevents damage to safety-related equipment and instrumentation located within containment.

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BASES (continued)

APPLICABILITY (continued)

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

In MODES 5 and 6, the probability and consequences of a LOCA or SLB are reduced due to the pressure and temperature limitations in these MODES. Therefore, the HMS is not required in these MODES to ensure containment OPERABILITY.

ACTIONS

A.1

With one train inoperable, the inoperable train must be restored to OPERABLE status within 30 days. The 30-day Completion Time is based on the availability of the second HMS train, the low probability of the occurrence of a LOCA or SLB that would generate hydrogen in amounts capable of exceeding the flammability limit, the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit, and the availability of the hydrogen recombiners, the Containment Spray System, the Hydrogen Purge System, and hydrogen monitors.

Concurrent failure of two HMS trains within a 30-day period is considered a low-probability event. If such a double failure were to occur, it would be indicative of poor HMS reliability and would result in the loss of functional capability. Therefore, LCO 3.0.3 must be immediately entered.

B.1

If an inoperable HMS train cannot be returned to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

Operating each HMS train for ≥ 15 minutes ensures that each train is OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. The 92-day Frequency is consistent with Inservice Inspection and Testing Program Surveillance Frequencies, operating experience, the known reliability of the fan motors and controls, and the two-train redundancy available.

SR 3.6.10.2

Demonstrating that each HMS train flow rate on slow speed is ≥ [4000] cfm ensures that each system is capable of maintaining localized hydrogen concentrations below the flammability limit. The 18-month Frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.10.3

This SR ensures that each HMS responds properly to a Containment Phase A isolation signal. The Surveillance shall verify each fan starts on slow speed from the non-operating condition, and each fan shifts to slow speed from fast operating condition. The 18-month Frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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BASES (continued)

REFERENCES

1. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission.

2. [Unit Name] FSAR, Section [ ], "[Title]."

DRAFT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.11 Hydrogen Ignition System (HIS)—MODES 1 & 2 (Ice Condenser)

BASES

BACKGROUND

The HIS supports containment OPERABILITY in post-accident environments by reducing the potential of breach of primary containment due to a hydrogen-oxygen reaction. The HIS is required by 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), to reduce the hydrogen concentration in the primary containment following a degraded core accident. The HIS must be capable of handling, without loss of containment OPERABILITY, an amount of hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the plenum volume).

The hydrogen ignitors ensure containment OPERABILITY and minimize challenges to safety equipment located within primary containment by limiting the temperatures and pressures that could be experienced from a hydrogen burn following a degraded core accident. Protection of primary containment OPERABILITY limits leakage of fission-product radioactivity from primary containment OPERABILITY to the environment. Loss of primary containment OPERABILITY could cause site-boundary doses to exceed values given in 10 CFR 100 (Ref. 3) or the NRC staff-approved licensing basis (e.g., specified fraction of 10 CFR 100 limits).

As a result of NRC rulemaking following the Three Mile Island accident, 10 CFR 50.44 (Ref. 1) was amended to require plants with ice condenser containments to install suitable hydrogen control systems that would accommodate an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water, without loss of containment OPERABILITY. The HIS provides this required capability. This requirement was placed on ice condenser plants because of their small containment volume and low design pressure (compared with pressurized water reactor dry containments). Calculations indicate that if hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water were to collect in the primary containment, the resulting hydrogen concentration

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BASES (continued)

BACKGROUND (continued)

would be far above the lower flammability limit such that, if ignited from a random ignition source, the resulting hydrogen burn would seriously challenge the primary containment OPERABILITY and OPERABILITY of safety systems in the primary containment.

The HIS is based on the concept of controlled ignition using thermal ignitors, designed to be capable of functioning in a post-accident environment, seismically supported, and capable of actuation from the control room. A total of [64] ignitors are distributed throughout the various regions of containment in which hydrogen could be released or to which it could flow in significant quantities. The ignitors are arranged in two independent trains such that each containment region has at least two ignitors, one from each train, controlled and powered redundantly so that ignition would occur in each region even if one train failed to energize.

When the HIS is initiated, the ignitor elements are energized and heat up to a surface temperature ≥ [1700]°F. At this temperature they ignite the hydrogen gas that is present in the airspace in the vicinity of the ignitor. The HIS depends on the dispersed location of the ignitors so that local pockets of hydrogen at increased concentrations would burn before reaching a hydrogen concentration significantly higher than the lower flammability limit. Hydrogen ignition in the vicinity of the ignitors is assumed to occur when the local hydrogen concentration reaches [8.0] volume percent (v/o) and results in [85]% of the hydrogen present being consumed.

APPLICABLE SAFETY ANALYSES

The HIS causes hydrogen in containment to burn in a controlled manner as it accumulates following a degraded core accident (Ref. 4). Burning occurs at the lower flammability concentration, where the resulting temperatures and pressures are relatively benign. Without the system, hydrogen could build up to higher concentrations that could result in a violent reaction if ignited by a random ignition source after such a buildup.

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BASES (continued)

APPLICABLE SAFETY ANALYSIS (continued)

The hydrogen ignitors are not included for mitigation of a Design Basis Accident (DBA) because an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water is far in excess of the hydrogen calculated for the limiting DBA loss-of-coolant accident (LOCA). The hydrogen ignitors, however, have been shown by probabilistic risk analysis to be a significant contributor to limiting the severity of accident sequences that are commonly found to dominate risk for plants with ice condenser containments. As such, the hydrogen ignitors are considered to be risk significant in accordance with the NRC Interim Policy Statement.

LCO

Two HIS subsystems must be OPERABLE with power from two independent safety-related power supplies.

[For this facility, an OPERABLE HIS subsystem constitute the following:]

[For this facility, OPERABLE hydrogen ignitors in the same containment region constitute the following:]

Operation with at least one HIS subsystem ensures that the hydrogen in containment can be burned in a controlled manner. Unavailability of both HIS trains could lead to hydrogen buildup to higher concentrations, which could result in a violent reaction if ignited. The reaction could take place fast enough to lead to high temperatures and overpressurization of primary containment and, as a result, breach primary containment or cause primary containment leakage rates above those assumed in the safety analyses. Damage to safety-related equipment located in primary containment could also occur.

[For this facility, the following support systems are required to be OPERABLE to ensure HIS OPERABILITY:]

[For this facility, those required support systems which upon their failure do not require declaring the HIS inoperable and their justification are as follows:]

(continued)

BASES (continued)

APPLICABILITY

Requiring OPERABILITY in MODES 1 and 2 for the HIS ensures its immediate availability after safety injection and scram actuated on a LOCA initiation. In the post-accident environment, the two HIS subsystems are required to control the hydrogen concentration within primary containment to near its flammability limit of 4.1 v/o assuming a worst-case single failure. This ensures primary containment OPERABILITY and prevents damage to safety-related equipment and instruments located within primary containment.

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

In MODES 5 and 6, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the HIS is not required to be OPERABLE in MODES 5 and 6 to ensure containment OPERABILITY.

ACTIONS

A.1

With one HIS subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. The 7-day Completion Time is based on the low probability of the occurrence of a degraded core event that would generate hydrogen in amounts equivalent to a metal-water reaction of 75% of the core cladding, the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit, the low probability of failure of the OPERABLE HIS subsystem, and the availability of other systems to control hydrogen concentration.

Concurrent failure of two HIS subsystems within a 7-day period is considered to be a low-probability event. If such double failure were to occur, it would be indicative of poor HIS reliability and would result in the loss of functional capability. Therefore, LCO 3.0.3 must be entered immediately.

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BASES (continued)

ACTIONS (continued)

B.1

With two hydrogen ignitors in one or more containment regions inoperable, verify that both hydrogen ignitors in each of two adjacent regions are not inoperable. If it is determined that the protection has been lost in two adjacent regions, then Condition C is entered. Required Action B.2 calls for the restoration of one hydrogen ignitor in each region to OPERABLE status within 7 days. The 7-day Completion Time is based on the same reasons given under Action A.1.

Therefore, the Note allows each containment region to be treated independently of the others, with a separate Completion Time.

C.1

The plant must be placed in a MODE in which the LCO does not apply if the HIS subsystem(s) cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.11.1

This SR confirms that ≥ [64] hydrogen ignitors can be successfully energized and assures that at least one of each redundant pair of ignitors is OPERABLE in each containment region. The ignitors are simple resistance elements. Therefore, energizing provides assurance of OPERABILITY. The allowance of up to [two] inoperable hydrogen ignitors is acceptable because, although one inoperable hydrogen ignitor in a region would compromise redundancy in that region, the containment regions are interconnected so that ignition in one region would cause burning to progress to the others (i.e., there is overlap in each hydrogen ignitor's effectiveness between regions). Per the Note, however, no two inoperable hydrogen ignitors may exist in the same containment region. The Frequency of 92 days is based on

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

the Inservice Testing Program requirements for determining equipment OPERABILITY, and has been shown to be acceptable through operating experience.

SR 3.6.11.2

A more detailed functional test is performed every 18 months to verify system OPERABILITY. Each glow plug is visually examined to ensure that it is clean, and that the electrical circuitry is energized. All ignitors (glow plugs), including normally inaccessible ignitors, are visually checked for a glow to verify that they are energized. Additionally, the surface temperature of each glow plug is measured to be ≥ [1700]°F to demonstrate that a temperature sufficient for ignition is achieved. The 18-month Frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the SR is performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed on the 18-month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.44, "Standards for Combustible Gas Control Systems in Light Water-Cooled Power Reactors."

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

3. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance."

4. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.12 Iodine Cleanup System (ICS) (Atmospheric & Subatmospheric)

BASES

BACKGROUND

The ICS is provided per GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to reduce the concentration and quality of fission products released to the containment atmosphere following postulated accident. The ICS would function together with the containment spray and cooling systems following a Design Basis Accident (DBA) to reduce the potential release of radioactive material, principally iodine, from the containment to within values specified in 10 CFR 100 (Ref. 2) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

The ICS consists of two 100%-capacity, separate and redundant trains. Each train includes a heater; cooling coils; prefilter; moisture separator; high efficiency particulate air (HEPA) filter; activated charcoal adsorber section for removal of radioiodines; and a fan. Ductwork, valves and/or dampers, and instrumentation also form part of the system. The moisture separators function to reduce the moisture content of the airstream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure sections of the main HEPA filter bank. The upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates filtered recirculation of the containment atmosphere following receipt of a safety injection signal. The system design is described in Reference 3.

The moisture separator is included for moisture (free water) removal from the gas stream. Heaters are used to heat the gas stream, which lowers the relative humidity. Continuous operation of each train for at least 10 hours per month with the heaters on reduces moisture buildup on the HEPA filters and adsorbers. Both the moisture separator and heater are important to the effectiveness of the charcoal adsorbers.

The primary purpose of the heaters is to ensure that the relative humidity of the airstream entering the charcoal adsorbers is maintained below 70%, which is consistent with the assigned iodine- and iodide-removal efficiencies as per Regulatory Guide 1.52.

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BASES (continued)

BACKGROUND (continued)

Two ICS trains are provided to meet the requirement for redundancy and independence. Each ICS train is powered from a separate ENGINEERED SAFETY FEATURE bus and is provided with a separate power panel and control panel. Essential service water (ESW) is required to supply cooling water to the cooling coils.

During normal operation, the Containment Cooling System is aligned to bypass the ICS HEPA filters and charcoal adsorbers. For ICS operation following a DBA, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers.

The ICS reduces the radioactive iodine content of the containment atmosphere following a DBA. In the event of a DBA, loss of the ICS could cause site-boundary doses, and control room operator doses to exceed the values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive iodine within containment are a loss-of-coolant accident (LOCA), a steam line break (SLB), or a rod ejection accident (REA). In the analysis for each of these accidents, it is assumed that adequate containment leak tightness is intact at event initiation and that potential leakage to the environment is controlled by the rate of containment leakage. Additionally, it is assumed that the amount of radioactive iodine released is limited by minimizing the amount of iodine present in the containment atmosphere.

The ICS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 4) assumes that only one train of the ICS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive iodine provided by the remaining one train of this filtration system.

The acceptance criteria applied to accidental releases of radioactive material to the environment are given in terms of the total radiation dose received by:

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

a. A member of the general public who remains at the exclusion-area boundary 2 two hours following the onset of the postulated fission-product release; or

b. A member of the general public who remains at the low-population-zone boundary for the duration of the accident.

The limits established in Reference 2 are a whole-body dose of 25 rem, a dose of 300 rem to the thyroid from iodine exposure, or both, or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The ICS satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

Two independent and redundant trains of the ICS are required to ensure that at least one is available, assuming a single failure coincident with a loss of offsite power. Total system failure could result in the atmospheric releases from the containment exceeding the limits in the event of a DBA.

The ICS is considered OPERABLE when the individual components necessary to maintain the ICS filtration are OPERABLE in both trains. A train is considered OPERABLE when:

a. Its associated fan is OPERABLE;

b. Its associated HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions;

c. Its associated heater, moisture separator, ductwork, valves, and dampers are OPERABLE;

d. The safety injection actuation instrumentation is OPERABLE;

e. ESW is available to the cooling coils;

f. Electric power is available from an ESF bus; and

g. Its associated SRs are met.

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BASES (continued)

LCO (continued)

[For this facility, the support systems required OPERABLE to ensure ICS OPERABILITY are as follows:]

[For this facility, those required support systems which upon their failure do not require declaring the ICS inoperable and their justification are as follows:]

APPLICABILITY

In MODES 1, 2, 3, and 4, iodine is a fission product that can be released from the fuel to the reactor coolant as a result of a DBA. The DBAs that can cause a failure of the fuel cladding are a LOCA, SLB, and REA. Because these accidents are considered credible accidents in MODES 1, 2, 3, and 4, containment must be OPERABLE to ensure that the offsite dose limits of Reference 2 are not exceeded. The ICS functions to limit the amount of iodine in the containment atmosphere following these DBAs, limiting the amount of free iodine available for leakage from containment. As such, the ICS must be OPERABLE during MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low due to the pressure and temperature limitations of these MODES. The ICS is not required in these MODES to remove iodine from the containment atmosphere.

ACTIONS

A.1

With one ICS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine-removal needs after a DBA. The 7-day Completion Time is based on consideration of such factors as:

a. The availability of the OPERABLE redundant ICS train;

b. The fact that even with no ICS train in operation, iodine would still be removed from the containment atmosphere through absorption by the Containment Spray System; and

c. The Completion Time is adequate to make most repairs.

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BASES (continued)

ACTIONS (continued)

Concurrent failure of two ICS trains within the 7-day period is considered a low-probability event. If such a double failure were to occur, it would be indicative of potential problems concerning ICS reliability and would result in the loss of functional capability. Therefore, LCO 3.0.3 must be entered immediately.

B.1 and B.2

If the ICS train cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.12.1

Operating each ICS train for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. For systems with heaters, operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating plants indicates that the 10-hour period is adequate for moisture elimination on the adsorbers and HEPA filters. The 31-day Frequency was developed considering the known reliability of fan motors and controls, the two train redundancy available, the iodine-removal capability of the containment Spray System independent of the ICS.

SR 3.6.12.2

The Ventilation Filter Testing Program (VFTP) (Specification 5.8.4.5) encompasses all the ICS filter tests consistent with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

use and following specific operations). Specific test Frequencies and additional information are discussed in detail in VFTP. The following tests are included:

a. Verification of the in-place (cold) penetration and bypass dioctyl phthalate (DOP) test leakage of each ICS train (the cold DOP test confirms the validity of the pre-installation hot DOP test and allows proper filter performance to be inferred);

b. Verification of the in-place penetration and bypass halogenated hydrocarbon refrigerant gas test leakage of each ICS train (this test determines that no bypass paths exist through or around the charcoal adsorber bed);

c. Verification of the methyl iodide penetration of a charcoal sample from each filter bed (this test verifies that the charcoal absorption capability is within required limits); and

d. Verification that the flow rate of each ICS train and the pressure drop across the combined prefilters, HEPA filters, and charcoal adsorber banks are within the required limits.

SR 3.6.12.3

The automatic startup test verifies that both trains of equipment start on receipt of an actual or simulated test signal. The 18-month Frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power.

Operating experience has shown that these components usually pass the SR when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the system equipment OPERABILITY is demonstrated on a 31-day Frequency by SR 3.6.12.1.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.12.4

The filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal operation and must reposition for accident operation to draw air through the filters. The 18-month Frequency is considered to be acceptable based on the damper reliability and design, the mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the 18-month SR.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants":

   General Design Criterion 41, "Containment Atmosphere Cleanup";

   General Design Criterion 42, "Inspection of Containment Atmosphere Cleanup Systems"; and

   General Design Criterion 43, "Testing of Containment Atmosphere Cleanup Systems."

2. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

3. [Unit Name] FSAR, Section [ ], "[Fission Product Removal and Control Systems]."

4. [Unit Name] FSAR, Section [ ], "[Accident Analysis]."

5. Regulatory Guide 1.52, (Rev. 2), "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water Cooled Nuclear Power Plants."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.13 Vacuum Relief Valves (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

BASES

BACKGROUND

The purpose of the vacuum relief valves is to ensure the OPERABILITY of the containment vessel against negative pressure (i.e., lower pressure inside than outside). Excessive negative pressure inside containment can occur if there is an inadvertent actuation of containment cooling features, such as the Containment Spray System. Multiple equipment failures and/or human errors are necessary to cause inadvertent actuation of these systems.

Ensuring containment OPERABILITY limits leakage of fission-product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause site-boundary doses, in the event of a Design Basis Accident (DBA), to exceed values given in 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

[For this facility, the characteristics of the vacuum relief valves and their locations in the containment pressure vessel are as follows:]

APPLICABLE SAFETY ANALYSES

Design of the vacuum relief valves involves calculating the effect of inadvertent actuation of containment cooling features, which can reduce the atmospheric temperature (and hence pressure) inside containment (Ref. 2). Conservative assumptions are used for all the relevant parameters in the calculation; for example, for the Containment Spray System, the minimum spray-water temperature, maximum initial containment temperature, maximum spray flow, all spray trains operating, etc. The resulting containment pressure versus time is calculated, including the effect of the opening of the vacuum relief valves when their negative pressure setpoint is reached. It is also assumed that one valve fails to open.

The containment was designed for an external pressure load equivalent to [-2.5] psig. The inadvertent actuation of the

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

containment cooling features was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was [-0.3] psig. This resulted in a minimum pressure inside containment of [-2.0] psig, which is less than the design load.

The vacuum relief valves must also perform the containment isolation function in a containment high-pressure event. For this reason, the system is designed to take the full containment positive design pressure and the environmental conditions (temperature, pressure, humidity, radiation, chemical attack, etc.) associated with the containment DBA.

The vacuum relief valves satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

The LCO establishes the minimum equipment required to accomplish the vacuum relief function following the inadvertent actuation of containment cooling features. Two vacuum relief valves are required to be OPERABLE to ensure that at least one is available, assuming the other valve fails to open.

[For this facility, an OPERABLE vacuum relief valve constitutes the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure vacuum relief valves OPERABILITY.

[For this facility, those required support systems which upon their failure do not require declaring the vacuum relief valves inoperable and their justification are as follows:]

APPLICABILITY

In MODES 1, 2, 3, and 4, the containment cooling features, such as the Containment Spray System, are required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside containment could occur whenever these systems are required to be OPERABLE due to inadvertent actuation of these systems. Therefore, the vacuum relief

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BASES (continued)

APPLICABILITY (continued)

valves are required to be OPERABLE in MODES 1, 2, 3, and 4 to mitigate the effects of inadvertent actuation of the Containment Spray System, Quench Spray (QS) System, or Containment Cooling System.

In MODES 5 and 6, the probability and consequences of a DBA are reduced due to the pressure and temperature limitation of these MODES. The Containment Spray System, QS System, and Containment Cooling System are not required to be OPERABLE in MODES 5 and 6. Therefore, maintaining OPERABLE vacuum relief valves is not required in MODE 5 or 6.

ACTIONS

A.1

When one of the required vacuum relief valves is inoperable, the inoperable valve must be restored to OPERABLE status within 4 hours. The specified time period is consistent with other LCOs for the loss of one valve of a containment isolation system and is a reasonable amount of time to effect many types of repairs.

Concurrent failure of two vacuum relief valves would result in the loss of functional capability. Therefore, LCO 3.0.3 must be entered immediately.

B.1 and B.2

If the vacuum relief valve cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.13.1

This SR cites the Inservice Inspection and Testing Program, which establishes the requirement that inservice inspection of the American Society of Mechanical Engineers (ASME) Code

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Class 1, 2, and 3 components and inservice testing of ASME Class 1, 2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda (Ref. 3). Therefore, SR Frequency is governed by the Inservice Inspection and Testing Program.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

3. ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," Section XI, American Society of Mechanical Engineers, New York.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.14 Shield Building Air Cleanup System (SBACS) (Dual and Ice Condenser)

BASES

BACKGROUND

The SBACS is required by 10 CFR 50 Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials that leak from the primary containment into the shield building (secondary containment) following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential release of radioactive material, principally iodine, to within values specified in 10 CFR 100 (Ref. 2) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

The (dual/ice condenser) containments have a secondary containment called the shield building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss-of-coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The SBACS establishes a negative pressure in the annulus between the Shield Building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. Shield building OPERABILITY is required to ensure retention of Primary containment leakage and proper operation of the SBACS.

The SBACS consists of two separate and redundant trains. Each train includes a heater, cooling coils, prefilter, moisture separators, high-efficiency particulate air (HEPA) filter, activated charcoal adsorber section for removal of radioiodines, and fan. Ductwork, valves, and/or dampers and instrumentation also form part of the system. The moisture separators function to reduce the moisture content of the airstream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank. Only the upstream HEPA filter and the charcoal adsorber section are

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BASES (continued)

BACKGROUND (continued)

credited in the analysis. The system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a safety injection signal. The system is described in Reference 3.

The prefilters remove any large particles in the air, and the moisture separators remove any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Heaters may be included for moisture removal on systems that operate in high humidity. Continuous operation of each train, for at least 10 hours per month, with heaters on, reduces moisture buildup on their HEPA filters and adsorbers. The cooling coils cool the air to keep the charcoal beds from becoming too hot due to absorption of fission product.

During normal operation, the Shield Building Cooling System is aligned to bypass the SBACS's HEPA filters and charcoal adsorbers. For SBACS operation following a DBA, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers.

The SBACS reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the SBACS could cause site-boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

[For this facility, the shield building access opening doors consist of the following:]

APPLICABLE SAFETY ANALYSES

The SBACS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 4) assumes that only one train of the SBACS 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. The amount of fission products available for release from containment is determined for a LOCA.

The modeled SBACS actuation in the safety analysis is based upon a worst-case response time associated with exceeding a

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

safety injection initiating signal. The total response time, from exceeding the signal setpoint to attaining the negative pressure of [ ] inch water gauge in the shield building, is [23] seconds. This response time is composed of signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting.

The SBACS satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

In the event of a DBA, one SBACS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two trains of the Shield Building Air Cleanup System must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure. Furthermore, all Shield Building access opening doors must be closed.

A 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;

c. Heater, cooling coils, moisture separators, ductwork, valves, and dampers are OPERABLE; and

d. SRs are met.

[For this facility, a closed shield building access opening door constitutes the following:]

[For this facility, the following support systems are required OPERABLE to ensure SBACS OPERABILITY:]

[For this facility, those required support systems which upon their failure do not require declaring the SBACS inoperable and their justification are as follows:]

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission-product release to containment that leaks to the shield building. The large-break LOCA, on which this system's design is based, is a full-power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decreases as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the filtration system is not required to be OPERABLE (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

ACTIONS

A.1

With one SBACS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine-removal needs after a DBA. The 7-day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SBACS train and the low probability of a DBA occurring during the period. The Completion Time is adequate to make most repairs.

Concurrent failure of two SBACS trains would result in the loss of functional capability. Therefore, LCO 3.0.3 must be entered immediately.

B.1

When one or more shield building access doors are inoperable, the inoperable door must be restored to OPERABLE status within 24 hours. [The 24-hour Completion Time is considered reasonable for this facility for the following reasons:]

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BASES (continued)

ACTIONS (continued)

Condition B includes a Note to provide clarification that each shield building access opening door is treated as an independent entity with an independent Completion Time.

An additional Note is included above Required Action B.1 that allows entry and exit through closed access opening doors for normal transit.

C.1 and C.2

If the SBACS train and all shield building access doors cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.14.1

Operating each SBACS train 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. For systems with heaters, operation with the heaters on (automatic heater cycling to maintain temperature) for $\geq$ 10 hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating plants indicates that the 10-hour period is adequate for moisture elimination on the adsorbers and HEPA filters. The 31-day Frequency was developed in consideration of the known reliability of fan motors and controls, the two-train redundancy available, and the iodine-removal capability of the Containment Spray System.

SR 3.6.14.2

The Ventilation Filter Testing Program (VFTP) (Specification 5.8.4.5) encompasses all the SBACS filter tests consistent with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing HEPA filter performance, charcoal

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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. The following tests are included:

a. Verification of the in-place (cold) penetration and bypass dioctyl phthalate (DOP) test leakage of each SBACS train (the cold DOP test confirms the validity of the preinstallation hot DOP test and allows proper filter performance to be inferred);

b. Verification of the in-place penetration and bypass halogenated hydrocarbon refrigerant gas test leakage of each SBACS train (this test determines that no bypass paths exist through or around the charcoal adsorber bed);

c. Verification of the methyl-iodide penetration of a charcoal sample from each filter bed (this test verifies that the charcoal adsorption capability is within required limits);

d. Verification that the flow rate of each SBACS train and the pressure drop across the combined prefilters, HEPA filters, and charcoal adsorber banks are within the required limits; and

e. Verification, for systems with heaters, of the proper function of each SBACS train's heaters.

SR 3.6.14.3

The automatic startup ensures that each SBACS train responds properly. The 14-month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unplanned plant transients if the SR is performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed on the 14-month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the SBACS

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

equipment OPERABILITY is demonstrated on a 31-day Frequency by SR 3.6.14.1.

SR 3.6.14.4

The filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal operation and must reposition for accident operation to draw air through the filters. The Frequency of 14 months is considered to be acceptable based on damper reliability and design, mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the 14-month SR.

SR 3.6.14.5

The proper functioning of the fans, dampers, filters, adsorbers, etc. as a system is verified by the ability to produce the required negative pressure ≥ 0.5-inch water gauge with respect to the adjacent area during test operation within one minute. The negative pressure assures that the building is adequately sealed and that leakage from the building will be prevented, since outside air will be drawn in by the low pressure. The negative pressure must be established within the time limit to ensure that no significant quantity of radioactive materials leak from the shield building prior to developing the negative pressure.

The Frequency of 14 months is consistent with Regulatory Guide 1.52 (Ref. 5) guidance for functional testing.

SR 3.6.14.6

[For this facility, the purpose of this SR is as follows:]

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

2. Title 10, Code of Federal Regulations, Part 100, 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

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BASES (continued)

REFERENCES (continued)

3. [Unit Name] FSAR, Section [ ], "[Fission Product Removal and Control Systems]."
4. [Unit Name] FSAR, Section [ ], "[Accident Analysis]."
5. Regulatory Guide 1.52, Rev. 2, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Absorption Units of Light-Water Cooled Nuclear Power Plants."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.15 Air Return System (ARS) (Ice Condenser)

BASES

BACKGROUND

The ARS is designed to assure the rapid return of air from the upper to the lower containment compartment after the initial blowdown following a Design Basis Accident (DBA). The return of this air to the lower compartment and subsequent recirculation back up through the ice condenser assists in cooling the containment atmosphere and limiting post-accident pressure and temperature in containment to less than design values. Limiting pressure and temperature would ensure containment OPERABILITY and reduce the release of fission-product radioactivity from containment to the environment in the event of a DBA to less than the guidelines of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

The ARS provides post-accident hydrogen mixing in selected areas of containment. The associated Hydrogen Skimmer System consists of hydrogen collection headers routed to potential hydrogen pockets in containment, terminating on the suction side of either of the two ARS fans at the header isolation valves. The minimum design flow from each potential hydrogen pocket is sufficient to limit the local concentration of hydrogen.

The ARS consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a 100%-capacity air return fan, associated damper, and hydrogen collection headers with isolation valves. Each train is powered from a separate ENGINEERED SAFETY FEATURE (ESF) bus.

The ARS fans are automatically started and the hydrogen collection header isolation valves are opened by the containment pressure High-High signal 10 minutes after the containment pressure reaches the pressure setpoint. The time delay ensures that no energy released during the initial phase of a DBA will bypass the ice bed through the ARS fans or Hydrogen Skimmer System.

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BASES (continued)

BACKGROUND (continued)

After starting, the fans displace air from the upper compartment to the lower compartment, thereby returning the air that was displaced by the high-energy line break blowdown from the lower compartment and equalizing pressures throughout containment. After discharge into the lower compartment, air flows with steam produced by residual heat through the ice condenser doors into the ice condenser compartment where the steam portion of the flow is condensed. The air flow returns to the upper compartment through the top deck doors in the upper portion of the ice condenser compartment. The ARS fans operate continuously after actuation, circulating air through the containment volume and purging all potential hydrogen pockets in containment. When the containment pressure falls below a predetermined value, the ARS fans are automatically de-energized. Thereafter, the fans are automatically cycled on and off if necessary to control any additional containment pressure transients.

The ARS also functions, after all the ice has melted, to circulate any steam still entering the lower compartment to the upper compartment where the Containment Spray System can cool it.

The ARS is an ESF system. It is designed to ensure that the heat-removal capability required during the post-accident period can be attained. The operation of the ARS in conjunction with the ice bed, the Containment Spray System, and the Residual Heat Removal (RHR) System spray, provide the required heat-removal capability to limit post-accident conditions to less than the containment design values.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment OPERABILITY are the loss-of-coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System, RHR System, and ARS being inoperable (Ref. 2). The DBA analyses show that the maximum peak containment

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

pressure results from the LOCA analysis and is calculated to be less than the containment design pressure.

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 System (ECCS) during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 3).

The maximum peak containment atmosphere temperature results from the SLB analysis and was calculated to exceed the containment design temperature for [unit-specific time period] during the transient. The basis of the containment design temperature, however, is to ensure OPERABILITY of safety-related equipment inside containment (Ref. 4). Thermal analyses show that the time interval during which the containment atmosphere temperature exceeds the containment design temperature is short enough that equipment surface temperatures remain below the design temperature. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB.

The analysis for minimum internal containment pressure (i.e., maximum external differential containment pressure) assumes inadvertent simultaneous actuation of both the ARS and the Containment Spray System. The containment vacuum relief valves are designed to accommodate inadvertent actuation of either or both systems.

The modeled ARS actuation from the containment OPERABILITY analysis is based upon a response time associated with exceeding the containment pressure High-High signal setpoint to achieving full ARS air flow. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The ARS total response time of 600 seconds consists of the built-in signal delay.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The ARS satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

In the event of a DBA, one train of the ARS with the Hydrogen Skimmer System is required to provide the minimum air recirculation for heat removal and hydrogen mixing assumed in the safety analyses. To ensure this requirement is met, two trains of the ARS with the Hydrogen Skimmer System must be OPERABLE. This will ensure that at least one train will operate assuming the worst-case single failure occurs, which is in the ESF power supply.

[For this facility, an OPERABLE ARS train constitutes the following:]

[For this facility, the following support systems are required OPERABLE to ensure ARS OPERABILITY:]

[For this facility, those required support systems which upon their failure do not require declaring the ARS inoperable and their justification are as follows:]

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ARS. Therefore, 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, the ARS is not required to be OPERABLE in these MODES to ensure containment OPERABILITY.

ACTIONS

A.1

If one of the required trains of the ARS is inoperable, it must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of

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BASES (continued)

ACTIONS (continued)

providing 100% of the heat-removal and hydrogen-skimming needs after an accident. The 72-hour Completion Time was developed taking into account the redundant heat-removal and hydrogen-skimming capability of the OPERABLE ARS train and the low probability of a DBA occurring in this period.

Concurrent failure of two ARSs would result in the loss of functional capability. Therefore, LCO 3.0.3. must be entered immediately.

B.1 and B.2

If the ARS train cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.15.1

Verifying that each ARS fan starts on an actual or simulated actuation signal and operates for ≥ 15 minutes is sufficient to ensure that all fans are OPERABLE and that all associated controls and time delays are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. The 92-day Frequency was developed considering the known reliability of fan motors and controls and the two-train redundancy available.

SR 3.6.15.2

Verifying fan motor current at rated speed with the return air dampers closed confirms one operating condition of the fan. This test is indicative of overall fan motor performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of 92 days conforms with the testing requirements for similar ESF equipment and considers the known reliability of

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

fan motors and controls and the two-train redundancy available.

SR 3.6.15.3

Verifying the OPERABILITY of the return air damper provides assurance that the proper flow path will exist when the fan is started. By applying the correct counterweight, the damper operation can be confirmed. The Frequency of 92 days was developed considering the importance of the dampers, their location, physical environment, and probability of failure. Operating experience has also shown this Frequency to be acceptable.

SR 3.6.15.4

Verifying the OPERABILITY of the motor-operated valve in the Hydrogen Skimmer System collection header to the lower containment compartment provides assurance that the proper flow path will exist when the valve receives an actuation signal. This Surveillance also confirms that the time delay to open is within specified tolerances. The 92-day Frequency was developed considering the known reliability of the motor-operated valves and controls and the two-train redundancy available. Operating experience has also shown this Frequency to be acceptable.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."
2. [Unit Name] FSAR, Section [ ], "[Containment Systems]."
3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."
4. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.16 Ice Bed (Ice Condenser)

BASES

BACKGROUND

The ice bed consists of over 2,721,600 lb of ice stored in baskets within the ice condenser. Its primary purpose is to provide a large heat sink in the event of a release of energy from a Design Basis Accident (DBA) in containment. The ice would absorb energy and limit containment peak pressure and temperature during the accident transient. Limiting the pressure and temperature would ensure containment OPERABILITY and reduce the release of fission-product radioactivity from containment to the environment to less than the guidelines of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

The ice condenser is an annular compartment enclosing approximately 300 degrees of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment which, for normal plant operation, are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper compartment, which also remain closed during normal plant operation. Intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. These doors also remain closed during normal plant operation. The upper plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice baskets held in the ice bed within the ice condenser are arranged to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes

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BASES (continued)

BACKGROUND (continued)

the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condenser limits the pressure and temperature buildup in containment. A divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a DBA and the additional heat loads that would enter containment during several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post-blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam from the lower compartment through the ice condenser where the heat is removed by the remaining ice.

As ice melts, the water passes through the ice condenser floor drains into the lower compartment. Thus, a second function of the ice bed is to be a large source of borated water (via the containment sump) for long-term Emergency Core Cooling System (ECCS) and Containment Spray System heat-removal functions in the recirculation mode.

A third function of the ice bed and melted ice is to remove fission-product iodine that may be released from the core during a DBA. Iodine removal occurs during the ice melt phase of the accident and continues as the melted ice is sprayed into the containment atmosphere by the Containment Spray System. The ice is adjusted to an alkaline pH that facilitates removal of radioactive iodine from the containment atmosphere. The alkaline pH also minimizes the occurrence of the chloride and caustic-stress corrosion on mechanical systems and components exposed to ECCS and Containment Spray System fluids in the recirculation mode of operation.

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BASES (continued)

BACKGROUND (continued)

It is important for the ice to be uniformly distributed around the 24 ice condenser bays and for open flow paths to exist around ice baskets. This is especially important during the initial blowdown so that the steam and water mixture entering the lower compartment do not pass through only part of the ice condenser, depleting the ice there while bypassing the ice in other bays.

Two phenomena that can degrade the ice bed during the long service period are:

a. Loss of ice by melting or sublimation; and

b. Obstruction of flow passages through the ice bed due to buildup of frost or ice. Both of these degrading phenomena are reduced by minimizing air leakage into and out of the ice condenser.

The ice bed ensures containment OPERABILITY by limiting the temperature and pressure that could be expected following a DBA. Ensuring containment OPERABILITY limits leakage of fission-product radioactivity from containment to the environment. In the event of a DBA, loss of containment OPERABILITY could cause site-boundary doses to exceed values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The ice bed ensures containment OPERABILITY by limiting the pressure and temperature that could be experienced following a DBA. The limiting DBAs considered relative to containment OPERABILITY are the loss-of-coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are not assumed to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the containment Spray System and the ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed in regards to containment ENGINEERED SAFETY FEATURE (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

failure and results in one train each of the Containment Spray System and ARS being inoperable.

The limiting DBA analyses (Ref. 2) show that the maximum peak containment pressure results from the LOCA analysis, and is calculated to be less than the containment design pressure.

For certain aspects of the transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 3).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for Specification 3.6.5, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The ice bed satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The ice bed LCO requires the existence of the required quantity of stored ice, appropriate distribution of the ice and the ice bed, open flow paths through the ice bed, and appropriate chemical content and pH of the stored ice. The stored ice functions to absorb heat during a DBA, thereby limiting containment air temperature and pressure. The chemical content and pH of the ice provide core shutdown margin (boron content) and remove radioactive iodine from the containment atmosphere when the melted ice is

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BASES (continued)

LCO (continued)

recirculated through the ECCS and the Containment Spray System, respectively.

Ice bed OPERABILITY comprises the following:

a. The ice bed temperature maintained well below the melting temperature to limit ice loss by melting and sublimation (SR 3.6.16.1).

[For this facility an OPERABLE ice bed temperature channel is established in LCO [ ] or SR [ ] and constitutes the following:]

b. A stored ice boron concentration of a least [1800] ppm as sodium tetraborate in order to meet the requirement for borated water when the melted ice is used in the ECCS recirculation mode of operation (SR 3.6.16.5.a). Sodium tetraborate is used since it has proven effective in maintaining the boron content for long storage periods. Sodium tetraborate also enhances the ability of the solution to remove and retain fission-product iodine.

c. A high pH, ≥ [9.0] and ≤ [9.5], to enhance the effectiveness of the ice and the melted ice in removing fission-product iodine from the containment atmosphere. The pH range also minimizes the occurrence of chloride and caustic-stress corrosion on mechanical systems and components exposed to the ECCS and Containment Spray System fluids in the recirculation mode of operation (SR 3.6.16.5.2).

d. A minimum ice mass of [2,721,600] lb, which includes a [27]% allowance above the mass assumed in the safety analyses. This allowance accounts for uncertainties in measurement and possible ice losses between Surveillances (SR 3.6.16.2).

e. The azimuthal distribution, which assures that the ice is uniformly distributed around the ice condenser bays. This is especially important during the initial blowdown so that the steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays (SR 3.6.16.3).

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BASES (continued)

LCO (continued)

f. Open flow channels through the ice bed and the ice condenser structures to ensure that the steam and lower containment atmosphere will pass through and contact the ice (SR 3.6.16.4); the partial blockage (> [0.38] inch) of more than one flow channel in a given ice condenser bay renders the ice bed inoperable.

g. Periodic visual inspections to identify any detrimental structural wear, cracks, corrosion or other damage that could impair the function of the ice baskets (SR 3.6.16.6).

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice bed. Therefore, 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, the ice bed is not required to be OPERABLE in these MODES to ensure containment OPERABILITY.

ACTIONS

A.1

If the ice bed is inoperable, it must be restored to OPERABLE status within 48 hours. The Completion Time was developed based on operating experience, which confirms that due to the very large mass of stored ice, the parameters comprising OPERABILITY do not change appreciably in this time period. Because of this fact, the Surveillance Frequencies are long (months), except for the ice bed temperature, which is checked every 12 hours. If a degraded condition is identified, even for temperature, with such a large mass of ice it is not possible for the degraded condition to significantly degrade further in a 48-hour period. Therefore, 48 hours is a reasonable amount of time to allow to correct a degraded condition before initiating a shutdown.

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BASES (continued)

ACTIONS (continued)

In the event that the required ice bed temperature channels are found inoperable, the ice bed is considered to be not within limits and Required Action A.1 applies.

B.1 and B.2

If the ice bed cannot be restored within limits in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.16.1

Verifying that the maximum temperature of the ice bed is ≤ [27]°F ensures that the ice is kept well below the melting point. The 12-hour Frequency was based on operating experience, which confirmed that, due to the large mass of stored ice, it is not possible for the ice bed temperature to degrade significantly in a 12-hour period, and on assessing the proximity of the LCO limit to the melting temperature.

Furthermore, the 12-hour Frequency is considered adequate in view of indications in the control room, including the alarm, to alert the operator to an abnormal ice bed temperature condition. This SR may be satisfied by use of the Ice Bed Temperature Monitoring System.

The required number of OPERABLE channels is established in LCO [ ] or SR [ ].

SR 3.6.16.2

The weighing program is designed to obtain a representative sample of the ice baskets. The representative sample shall include six baskets from each of the 24 ice condenser bays, and consist of one basket from radial rows 1, 2, 4, 6, 8, and 9. If no basket from a designated row can be obtained for weighing, a basket from the same row of an adjacent bay shall be weighed.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The rows chosen include the rows nearest the inside and outside walls of the ice condenser (rows 1 and 2, and 8 and 9, respectively), where heat transfer into the ice condenser is most likely to influence melting or sublimation. Verifying the total weight of ice ensures that there is adequate ice to absorb the required amount of energy to mitigate the DBAs.

If a basket is found to contain < [1400] lb of ice, a representative sample of 16 additional baskets from the same bay shall be weighed. The average weight of ice in these 21 baskets shall be ≥ [1400] lb at a 95% confidence level.

Weighing 16 additional baskets from the same bay in the event a Surveillance reveals that a single basket contains < [1400] lb ensures that no local zone exists that is grossly deficient in ice. Such a zone could experience early melt-out during a DBA transient, creating a path for steam to pass through the ice bed without begin condensed. The Frequency of 9 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses. Operating experience has verified that, with the 9-month Frequency, the weight requirements are maintained with no significant degradation between Surveillances.

SR 3.6.16.3

This SR ensures that the azimuthal distribution of ice is reasonably uniform, by verifying that the average ice weight in each of three azimuthal groups of ice condenser bays is within the limit. The Frequency of 9 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses. Operating experience has verified that, with the 9-month Frequency, the weight requirements are maintained with no significant degradation between Surveillances.

SR 3.6.16.4

This SR ensures that the flow channels through the ice condenser have not accumulated an excessive amount of ice or frost blockage. The visual inspection shall be made for two or more flow channels per ice condenser bay and shall

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

include the following specific locations along the flow channel:

a. Past the lower inlet plenum support structures and turning vanes;

b. Between ice baskets;

c. Past lattice frames;

d. Through the intermediate floor grating; and

e. Through the top deck floor grating.

The allowable [0.38]-inch-thick buildup of frost or ice is based on analysis of containment response to a DBA with partial blockage of the ice condenser flow passages. If a flow channel in a given bay is found to have an accumulation of frost or ice greater than [0.38] inches thick, a representative sample of 16 additional flow channels from the same bay shall be visually inspected.

If these additional flow channels are all found to be acceptable, the discrepant flow channel may be considered single, unique, and acceptable deficiency. More than one discrepant flow channel in a bay is not acceptable, however. These requirements are based on the sensitivity of the partial blockage analysis to additional blockage. The Frequency of 9 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses.

SR 3.6.16.5

Verifying the chemical composition of the stored ice ensures that the stored ice has a boron concentration of at least 1800 ppm as sodium tetraborate and a high pH, $\geq$ [9.0] and $\leq$ [9.5], in order to meet the requirement for borated water when the melted ice is used in the ECCS recirculation mode of operation. Sodium tetraborate has been proven effective in maintaining the boron content for long storage periods, and it also enhances the ability of the solution to remove and retain fission-product iodine. The high pH is required to enhance the effectiveness of the ice and the melted ice in removing iodine from the containment atmosphere. This pH

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

range also minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to ECCS and Containment Spray System fluids in the recirculation mode of operation. The Frequency of 18 months was developed considering these facts:

a. Long-term ice storage tests have determined that the chemical composition of the stored ice is extremely stable;

b. Operating experience has demonstrated that meeting the boron concentration and pH requirements has never been a problem; and

c. Someone would have to enter the containment to take the sample, and if the unit is at power, that person would receive a radiation dose.

SR 3.6.16.6

This SR ensures that a representative sampling of ice baskets, which are relatively thin-walled perforated cylinders, have not been degraded by wear, cracks, corrosion, or other damage. Each ice basket shall be raised at least 12 feet for this inspection. The Surveillance Frequency of 40 months for a visual inspection of the structural soundness of the ice baskets is based on engineering judgment, and considers such factors as the thickness of the basket walls relative to corrosion rates expected in their service environment and the results of the long-term ice storage testing.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.17 Ice Condenser Doors (Ice Condenser)

BASES

BACKGROUND

The ice condenser doors consist of the inlet doors, the intermediate deck doors, and the top deck doors. The functions of the doors are to:

a. Seal the ice condenser from air leakage during the lifetime of the unit; and

b. Open in the event of a Design Basis Accident (DBA) to direct the hot steam-air mixture from the DBA into the ice Bed, where the ice would absorb energy and limit Containment peak pressure and temperature during the accident transient.

Limiting the pressure and temperature following a DBA would ensure containment OPERABILITY and reduce the release of fission-product radioactivity from containment to the environment to less than the guidelines of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

The ice condenser is an annular compartment enclosing approximately 300° of the perimeter of the upper containment compartment, but penetrating the operating deck so a portion extends into the lower containment compartment. The inlet doors separate the atmosphere of the lower compartment from the ice bed inside the ice condenser. The top deck doors are above the ice bed and exposed to the atmosphere of the upper compartment. The intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. This plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice baskets held in the ice bed within the ice condenser are arranged to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

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BASES (continued)

BACKGROUND (continued)

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condensers limits the pressure and temperature buildup in containment. A divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

The ice, together with the containment spray, serves as a containment heat-removal system and is adequate to absorb the initial blowdown of steam and water from a DBA as well as the additional heat loads that would enter containment during the several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post-blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment, and to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice.

The water from the melted ice drains into the lower compartment where it serves as a source of borated water (via the containment sump) for the Emergency Core Cooling System (ECCS) and the Containment Spray System heat-removal functions in the recirculation mode. The ice (via the Containment Spray System) and the recirculated ice melt also serve to clean up the containment atmosphere.

The ice condenser doors assure that the ice stored in the ice bed is preserved during normal operation (doors closed), and that the ice condenser functions as designed if called upon to act as a passive heat sink following a DBA. As such, the ice condenser doors are a contributing factor in ensuring containment OPERABILITY. Ensuring containment OPERABILITY limits leakage of fission-product radioactivity

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BASES (continued)

BACKGROUND (continued)

from containment to the environment. Loss of containment OPERABILITY could cause site-boundary doses, in the event of a DBA, to exceed values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment OPERABILITY are the loss-of-coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed with respect to ENGINEERED SAFETY FEATURE (ESF) systems, assuming the loss of one ESF bus, which is the worst-case single active failure and results in one train each of the Containment Spray System and the ARS being rendered inoperable.

The limiting DBA analyses (Ref. 2) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS's cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 3).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in Bases B 3.6.5B, "Containment Air Temperature."

An additional design requirement was imposed on the ice condenser door design for a small-break accident in which the flow of heated air and steam is not sufficient to fully open the doors.

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APPLICABLE SAFETY ANALYSES (continued)

For this situation, the doors are designed so that all of the doors would partially open by approximately the same amount. Thus, the partially opened doors would modulate the flow so that each ice bay would receive an approximately equal fraction of the total flow.

[For this facility, partial opening of the ice condenser doors is accomplished as follows:]

This design feature assures that the heated air and steam will not flow preferentially to some ice bays and deplete the ice there without utilizing the ice in the other bays.

In addition to calculating the overall peak containment pressures, the DBA analyses include the calculation of the transient differential pressures that would occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand the local transient pressure differentials for the limiting DBAs.

The ice condenser doors satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

This LCO establishes the minimum equipment requirements to assure that the ice condenser doors perform their safety function. The ice condenser inlet doors, intermediate deck doors, and top deck doors must be closed to minimize air leakage into and out of the ice condenser, with its attendant leakage of heat into the ice condenser and loss of ice through melting and sublimation. The doors must be OPERABLE to ensure the proper opening of the ice condenser in the event of a DBA. OPERABILITY includes being free of any obstructions that would limit their opening, and for the inlet doors, being adjusted such that the opening and closing torques are within limits. The ice condenser doors ensure containment OPERABILITY by functioning with the ice condenser to limit the pressure and temperature that could be expected following a DBA.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice condenser doors. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

The probability and consequences of these events in MODES 5 and 6 are reduced due to the pressure and temperature limitation of these MODES. Therefore, the ice condenser doors are not required to be OPERABLE in these MODES to ensure containment OPERABILITY.

A Note has been added to provide clarification that for this LCO, all ice condenser doors are treated as an entity with a single Completion Time.

ACTIONS

A.1

If one or more inlet doors are physically restrained from opening, the door(s) must be restored to OPERABLE status 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 containment to be restored to OPERABLE status in 1 hour.

B.1

If one or more doors are determined to be partially open or otherwise inoperable for reasons other than Condition A, or a door is found that is not closed, it is acceptable to continue unit operation for up to 14 days provided the ice bed temperature instrumentation is monitored once per 4 hours to ensure that the open or inoperable door is not allowing enough air leakage to cause the maximum ice bed temperature to approach the melting point. The Frequency of 4 hours is based on the fact that temperature changes cannot occur rapidly in the ice bed because of the large mass of ice involved. The 14-day Completion Time is based on long-term ice storage tests that indicate that if the temperature is maintained below [27]°F, there would not be a significant loss of ice from sublimation. If the maximum ice bed temperature is > [27]°F at any time, the situation reverts to Condition C and a Completion Time of 48 hours is

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BASES (continued)

ACTIONS (continued)

allowed to restore the inoperable door to OPERABLE status or enter into Required Actions D.1 and D.2. ice bed temperature must be verified to be within the specified Frequency as augmented by the provisions of SR 3.0.2. If this verification is not made, Required Action D.1 and Required Action D.2, not Required Action C.1, must be taken.

C.1

If the Required Actions B.1 or B.2 are not met, the doors must be restored to OPERABLE status within 48 hours. The 48-hour Completion Time is based on the fact that, with the very large mass of ice involved, it would not be possible for the temperature to decrease to the melting point and a significant amount of ice to melt in a 48-hour period. Condition C is entered from Condition B only when the Completion Time of Required Action B.2 is not met or the ice Bed temperature has not been verified at the required Frequency.

D.1 and D.2

If the ice condenser doors cannot be restored to within limits within the associated Completion Time, the plant must be placed in MODE in which the LCO does not apply. This is done by placing the plant 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 MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.17.1

Verifying, by means of the Inlet Door Position Monitoring System, that the inlet doors are in their closed positions makes the operator aware of an inadvertent opening of one or more doors. The Surveillance Frequency of 12 hours ensures that operators on each shift are aware of the status of the doors.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.17.2

Verifying by visual inspection that each intermediate deck door is closed and not impaired by ice, frost, or debris provides assurance that the intermediate deck doors (which form the floor of the upper plenum where frequent maintenance on the ice bed is performed) have not been left open or obstructed. The Frequency of 7 days is based on engineering judgment and takes into consideration such factors as the frequency of entry into the intermediate ice condenser deck, the time required for significant frost buildup, and the probability that a DBA will occur.

SR 3.6.17.3

Verifying by visual inspection that the ice condenser inlet doors are not impaired by ice, frost, or debris provides assurance that the doors are free to open in the event of a DBA. For this facility, the Frequency of [6] months (3 months during the first year after receipt of license) is based on door design, which does not allow water condensation to freeze, and operating experience, which indicates that the inlet doors very rarely fail to meet their SR acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown.

SR 3.6.17.4

Verifying the opening torque of the inlet doors provides assurance that no doors have become stuck in the closed position. The value of [675] inch·lb is based on the design opening pressure on the doors of 1.0 lb/ft^2. For this facility, the Frequency of [6] months (3 months during the first year after receipt of license) is based on the passive nature of the closing mechanism (i.e., once adjusted, there are no factors that would change the setting, except possibly a buildup of ice; ice buildup is not likely, however, because of the door design, which does not allow water condensation to freeze). Operating experience indicates that the inlet doors very rarely fail to meet their SR acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.17.5

The torque test Surveillance ensures that the inlet doors have not developed excessive friction and that the return springs are producing a door-return torque within limits. The torque test consists of the following:

1. Verify that the torque, T(OPEN), required to cause opening motion at the [40]° open position, is ≤ [195] inch·lb;

2. Verify that the torque, T(CLOSE), required to hold the door stationary (i.e., keep it from closing) at the [40]° open position, is ≥ [78] inch·lb; and

3. Calculate the frictional ≥ torque, T(FRICT) = 0.5 {T(OPEN) - T(CLOSE)}, and verify that the T(FRICT) is ≤ [40] inch·lb.

The purpose of the friction and return torque specifications is to ensure that, in the event of a small-break LOCA or SLB, all of the 24 door pairs open uniformly. This assures that, during the initial blowdown phase, the steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays. For this facility, the Frequency of [6] months (3 months during the first year after receipt of license) is based on the passive nature of the closing mechanism (i.e., once adjusted, there are no known factors that would change the setting, except possibly a buildup of ice; ice buildup is not likely, however, because of the door design, which does not allow water condensation to freeze). Operating experience indicates that the inlet doors very rarely fail to meet their SR acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown.

SR 3.6.17.6

Verifying the OPERABILITY of the intermediate deck doors provides assurance that the intermediate deck doors are free to open in the event of a DBA. The verification consists of

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

visually inspecting the intermediate doors for structural deterioration, verifying free movement of the vent assemblies, and ascertaining free movement of each door when lifted with the applicable force shown below:

	Door	Lifting Force
a.	Adjacent to crane wall	≤ 37.4 lb
b.	Paired with door adjacent to crane wall	≤ 33.8 lb
c.	Adjacent to containment wall	≤ 31.8 lb
d.	Paired with door adjacent to containment wall	≤ 31.0 lb

The 18-month Frequency (3 months during the first year after receipt of license) is based on the passive design of the intermediate deck doors, the frequency of personnel entry into the intermediate deck, and the fact that SR 3.6.17.2 confirms on a 7-day Frequency that the doors are not impaired by ice, frost, or debris, which are ways a door would fail the opening-force test (i.e., by sticking or from increased door weight).

SR 3.6.17.7

Verifying by visual inspection that the top desk doors are in place and not obstructed provides assurance that the doors are performing their function of keeping warm air out of the ice condenser during normal operation, and would not be obstructed if called upon to open in response to a DBA. The Frequency of 92 days is based on engineering judgment, which considered such factors as the following:

a. The relative inaccessibility and lack of traffic in the vicinity of the doors make it unlikely that a door would be inadvertently left open;

b. Excessive air leakage would be detected by temperature monitoring in the ice condenser; and

c. The light construction of the doors would assure that, in the event of a DBA, air and gases passing through the ice condenser would find a flow path, even if a door were obstructed.

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BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. [Unit Name] FSAR, Section [ ], "[Title]."

3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

3.6 CONTAINMENT SYSTEMS

B 3.6.18 Divider Barrier Integrity (Ice Condenser)

BASES

BACKGROUND

The divider barrier consists of the operating deck and associated seals, personnel access doors, and equipment hatches that separate the upper and lower containment compartments. Divider barrier integrity is necessary to minimize bypassing of the ice condenser by the hot steam and air mixture released into the lower compartment during a Design Basis Accident (DBA). This ensures that most of the gases pass through the ice Bed, which condenses the steam and limits pressure and temperature during the accident transient. Limiting the pressure and temperature would ensure containment OPERABILITY and reduce the release of fission-product radioactivity from containment to the environment in the event of a DBA to less than the guidelines of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the door panels at the top of the condenser to open, which allows the air to flow out of the ice condenser into the upper compartment. The ice condenses the steam as it enters, thus limiting the pressure and temperature buildup in containment. The divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser. The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a DBA as well as the additional heat loads that would enter containment over several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post-blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in

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BASES (continued)

BACKGROUND (continued)	containment and to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice. Divider barrier integrity ensures that the high-energy fluids released during a DBA would be directed through the ice condenser, and that the ice condenser would function as designed if called upon to act as a passive heat sink following a DBA. As such, the divider barrier ensures containment OPERABILITY by limiting the pressure and temperature that could be expected following a DBA. Ensuring containment OPERABILITY limits leakage of fission-product radioactivity from containment OPERABILITY to the environment. Loss of containment OPERABILITY could cause site-boundary doses, in the event of a DBA, to exceed values given in the licensing basis.
APPLICABLE SAFETY ANALYSES	Divider barrier integrity ensures the functioning of the ice condenser to the limiting containment pressure and temperature that could be experienced following a DBA. The limiting DBAs considered relative to containment OPERABILITY are the loss-of-coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and the ARS also function to assist the ice Bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed, with respect to containment ENGINEERED SAFETY FEATURE (ESF) systems, assuming the loss of one ESF bus, which is the worst-case single active failure and results in the inoperability of one train in both the Containment Spray System and the ARS. The limiting DBA analyses (Ref. 2) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum peak containment temperature results from the SLB analysis and is discussed in Bases B 3.6.5B, "Containment Air Temperature."

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The divider barrier satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

This LCO establishes the minimum equipment requirements to ensure that the divider barrier performs its safety function of ensuring that bypass leakage, in the event of a DBA, does not exceed the bypass leakage assumed in the accident analysis. Included are the requirements that the personnel access doors and equipment hatches in the divider barrier are OPERABLE and closed, and that the divider barrier seal is properly installed and has not degraded with time. An exception to the requirement that the doors be closed is made to allow personnel transit entry through the divider barrier. The basis of this exception is the assumption that, for personnel transit, the time during which a door is open will be short (i.e., shorter than the Completion Time of 1 hour for Condition A). The divider barrier ensures containment OPERABILITY by functioning with the ice condenser to limit the pressure and temperature that could be expected following a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the integrity of the divider barrier. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

The probability and consequences of these events in MODES 5 and 6 are low due to the pressure and temperature limitations of these MODES. As such, divider barrier integrity is not required in these MODES to ensure containment OPERABILITY.

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BASES (continued)

ACTIONS

A.1

If one or more personnel access doors or equipment hatches is inoperable or open, except for personnel transit entry, 1 hour is allowed to restore the door(s) to OPERABLE status and the closed position. The 1-hour Completion Time is consistent with LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Required Action A.1 has been modified by a Note to provide clarification that for this LCO, all personnel access doors and equipment hatches are treated as an entity with a single Completion Time.

B.1

If the divider barrier seal is inoperable, 1 hour is allowed to restore the seal to OPERABLE status. The 1-hour Completion Time is consistent with LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

C.1 and C.2

If the Required Actions are not met within the required Completion Time, the plant must be placed in a MODE where the requirement does not apply. This is performed by placing the plant in MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.18.1

Verification by visual inspection that all personnel access doors and equipment hatches between the upper and lower containment compartments are closed provides assurance that divider barrier integrity is maintained prior to the reactor being taken from MODE 5 to MODE 4. This SR is necessary because many of the doors and hatches may have been opened for maintenance during the shutdown.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.18.2

Verification that the personnel access door and equipment-hatch seals, sealing surfaces, and alignments are acceptable provides assurance that divider barrier integrity is maintained. This inspection cannot be made when the door or hatch is closed. Therefore, SR 3.6.18.2 is required for each door or hatch that has been opened, prior to the final closure. Some doors and hatches may not be opened for long periods of time. Those that use resilient materials in the seals must be opened and inspected at least once every 10 years to provide assurance that the seal material has not aged to the point of degraded performance. The Frequency of 10 years is based on the known resiliency of the materials used for seals, the fact that the openings have not been opened (to cause wear), and operating experience that confirms that the seals inspected at this Frequency have been found to be acceptable.

SR 3.6.18.3

Verification after each opening of a personnel access door or equipment hatch that it has been closed makes the operator aware of the importance for closing it and thereby provides additional assurance that divider barrier integrity is maintained while in applicable MODES.

SR 3.6.18.4

Conducting periodic physical property tests on divider barrier seal test coupons provides assurance that the seal material has not degraded in the containment environment, including the effects of irradiation with the reactor at power. The required tests include a tensile-strength test and a test for elongation. The Frequency of 18 months was developed considering such factors as the known resiliency of the seal material used, the inaccessibility of the seals and absence of traffic in their vicinity, and the plant conditions needed to perform the SR. Operating experience has shown that these components usually pass the Surveillance test when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.18.5

Visual inspection of the seal around the perimeter provides assurance that the seal is properly secured in place. The Frequency of 18 months was developed considering such factors as the inaccessibility of the seals and absence of traffic in their vicinity, the strength of the bolts and mechanisms used to secure the seal, and the plant conditions needed to perform the SR. Operating experience has shown that these components usually pass the Surveillance test when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."
2. [Unit Name] FSAR, Section [ ], "[Containment Systems]."
3. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.19 Containment Recirculation Drains (Ice Condenser)

BASES

BACKGROUND

The containment recirculation drains consist of the ice condenser drains and the refueling canal drains. The ice condenser is partitioned into 24 bays, each having a pair of inlet doors that open from the bottom plenum to allow the hot steam/air mixture from a Design Basis Accident (DBA) to enter the ice condenser. Twenty of the 24 bays have an ice condenser floor drain at the bottom to drain the melted ice into the lower compartment (in the 4 bays that do not have drains, the water drains through the floor drains in the adjacent bays). Each drain leads to a drain pipe that drops down several feet, then makes one or more 90° bends and exits into the lower compartment. A check (flapper) valve at the end of each pipe keeps warm air from entering during normal operation, but when the water exerts pressure, it opens to allow the water to spill into the lower compartment. This prevents water from backing up and interfering with the ice condenser inlet doors. The water delivered to the lower containment serves to cool the atmosphere as it falls through to the floor and provide a source of borated water at the containment sump for long-term use by the Emergency Core Cooling System (ECCS) and the containment Spray System during the recirculation MODE of operation.

The two refueling canal drains are at low points in the refueling canal. During a refueling, plugs are installed in the drains and the canal is flooded to facilitate the refueling process. The water acts to shield and cool the spent fuel as it is transferred from the reactor vessel to storage. After refueling, the canal is drained and the plugs removed. In the event of a DBA, the refueling canal drains are the main return path to the lower compartment for Containment Spray System water sprayed into the upper compartment.

The ice condenser drains and the refueling canal drains ensure containment OPERABILITY by functioning with the ice bed, the Containment Spray System, and the ECCS to limit the pressure and temperature that could be expected following a DBA.

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BASES (continued)

BACKGROUND (continued)	Ensuring containment OPERABILITY limits leakage of fission-product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause site-boundary doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).
APPLICABLE SAFETY ANALYSES	The containment recirculation drains protect the integrity of the containment by functioning with the ice condenser, the Containment Spray System, and the ECCS to limit the pressure and temperature that could be expected following a DBA. The limiting DBAs considered relative to containment OPERABILITY are the loss-of-coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and the Air Return System (ARS) also function to assist the ice bed in limiting pressures and temperatures. Therefore, the analysis of the postulated DBAs, with respect to ENGINEERED SAFETY FEATURE (ESF) systems, assumes the loss of one ESF bus, which is the worst-case single active failure and results in one train of the Containment Spray System and one train of the ARS being rendered inoperable. The limiting DBA analyses (Ref. 2) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in Bases B 3.6.5, "Containment Air Temperature." In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs. The containment recirculation drains satisfy Criterion 3 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

LCO

This LCO establishes the minimum requirements to ensure that the containment recirculation drains perform their safety functions. The ice condenser floor-drain valve disks must be closed to minimize air leakage into and out of the ice condenser during normal operation and must open in the event of a DBA when water begins to drain out. The refueling canal drains must have their plugs removed and remain clear to ensure the return of Containment Spray System water to the lower containment in the event of a DBA. The containment recirculation drains ensure containment OPERABILITY by functioning with the ice condenser, ECCS, and Containment Spray System to limit the pressure and temperature that could be expected following a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature, which would require the operation of the containment recirculation drains. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

The probability and consequences of these events in MODES 5 and 6 are low due to the pressure and temperature limitations of these MODES. As such, the containment recirculation drains are not required to be OPERABLE in these MODES to ensure containment OPERABILITY.

ACTIONS

A.1

If one ice condenser floor drain is inoperable, 1 hour is allowed to restore the drain to OPERABLE status. 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 require that containment be restored to OPERABLE status within 1 hour.

Concurrent failure of more than one ice condenser Floor Drain will take the facility farther outside the bounds of the containment analysis. Therefore, LCO 3.0.3 must be immediately entered.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

B.1

If one refueling canal drain is inoperable, 1 hour is allowed to restore the drain to OPERABLE status. 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 require that containment be restored to OPERABLE status in 1 hour.

Concurrent failure of more than one refueling canal drain will take the facility farther outside the bounds of the containment analysis. Therefore, LCO 3.0.3 must be immediately entered.

C.1 and C.2

If the affected drain(s) cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.19.1

Verifying the OPERABILITY of the refueling canal drains provides assurance that they will be able to perform their functions in the event of a DBA. This Surveillance confirms that the refueling canal drain plugs have been removed and that the drains are clear of any obstructions that could impair their functioning. In addition to debris near the drains, attention must be given to any debris that is located where it could be moved to the drains in the event that the Containment Spray System is in operation and water is flowing to the drains. SR 3.6.19.1 must be performed before entering MODE 4 from MODE 5 after every filling of the canal to ensure that the plugs have been removed and that no debris that could impair the drains was deposited during the time the canal was filled. The 92-day Frequency

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

was developed considering such factors as the inaccessibility of the drains, the absence of traffic in the vicinity of the drains, and the redundancy of the drains.

SR 3.6.19.2

Verifying the OPERABILITY of the ice condenser floor drains provides assurance that they will be able to perform their functions in the event of a DBA. Inspecting the drain valve disk ensures that the valve is performing its function of sealing the drain line from warm air leakage into the ice condenser during normal operation, yet will open if melted ice fills the line following a DBA. Verifying that the drain lines are not obstructed ensures their readiness to drain water from the ice condenser. The 18-month Frequency was developed considering such factors as the inaccessibility of the drains during power operation; the design of the ice condenser, which precludes melting and refreezing of the ice; and operating experience that has confirmed that the drains are found to be acceptable when the Surveillance is performed on an 18-month Frequency. Because of high radiation in the vicinity of the drains during power operation, this Surveillance is normally done during a shutdown.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. [Unit Name] FSAR, Section [ ], "[Containment Systems]."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.20 Shield Building (Dual & Ice Condenser)

BASES

BACKGROUND

The shield building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects containment leakage that may occur following a loss-of-coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The Shield Building Air Cleanup System (SBACS) establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. The shield building is required to be OPERABLE to ensure retention of primary containment leakage and proper operation of the SBACS.

APPLICABLE SAFETY ANALYSES

The design basis for shield building OPERABILITY is a LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive materials from the primary containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analysis. This restriction, in conjunction with the operation of the SBACS, will limit the site-boundary radiation doses to within the limits of 10 CFR 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits) during an accident.

LCO

Shield building OPERABILITY must be maintained to ensure proper operation of the SBACS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analysis.

[For this facility, an OPERABLE shield building constitutes the following:]

(continued)

BASES (continued)

APPLICABILITY

Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the primary containment following a LOCA. Therefore, shield building OPERABILITY is required during the same operating conditions that require containment OPERABILITY. containment OPERABILITY and shield building OPERABILITY are required in MODES 1, 2, 3, and 4 when a steam line break, LOCA, or rod ejection accident could release radioactive material to the primary containment atmosphere.

In MODES 5 and 6, the probability and consequences of these Design Basis Accident (DBA) events are low due to the temperature and pressure limitations in these MODES. Therefore, shield building OPERABILITY is not required in MODE 5 or 6.

ACTIONS

A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours.

[For this facility, the 24-hour Completion Time is considered reasonable, based on the following:]

B.1 and B.2

If shield building OPERABILITY cannot be restored in the required time period, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant 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 MODES from full power in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.20.1

Verifying that shield building annulus 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 considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.

SR 3.6.20.2

Maintaining shield building OPERABILITY requires maintaining each door in the access opening closed, except when the access opening is being used for normal transient entry and exit (then at least one door must remain closed). The 31-day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.6.20.3

This SR would give advance indication of gross deterioration of the concrete structural integrity of the shield building. The Frequency of this SR is the same as that of SR 3.6.1.1. The verification is done during shutdown and as part of Type A leakage tests associated with SR 3.6.1.1.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in Reference 1. The MSSV capacity criteria is 110% of rated steam flow at 110% of the steam generator design pressure. This meets the requirements of Section III of the ASME Code (Ref. 2). The MSSV design includes staggered setpoints, as shown in Table 3.7.1-1, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine-reactor trip.

The valve lift settings given in Table 3.7.1-1 meet the requirements of Section III of the ASME Code (Ref. 2). The total relieving capacity for all [20] MSSVs at 110% of system design pressure (adjusted for a [50] psi pressure drop to valves inlet) is [19.44 E6] lb/hour. This capacity is less than the total rated capacity because the MSSVs operate at an inlet pressure below rated conditions, ensuring that steam generator pressure does not exceed 110% of design. At these same secondary pressure conditions, the total steam flow at [102]% of [3,424 Mwt (RATED THERMAL POWER (RTP) plus 14 Mwt pump heat input)] is [17.83 E6] lb/hour. The ratio of the total steam flow to the total steam flow capacity is [109.2]%.

The low-pressure setpoint MSSV, [1190] psia, corresponds to a zero-power loop average temperature (T_{avg}), secondary fluid saturation temperature of [556]°F. The RCS T_{avg} must be above this temperature to open the MSSVs.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from the ASME Code; its purpose is to limit the 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 operating occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the MSSVs' relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in Reference 3. Of these, the full power turbine trip without steam dump is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The transient response for turbine trip without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. If a minimum reactivity feedback is assumed, the reactor is tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure.

If maximum reactivity feedback is assumed, the reactor is tripped on overtemperature ΔT. The departure from nucleate boiling ratio increases throughout the transient, and never drops below its initial value. Pressurizer relief valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems. The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

The accident analysis requires two MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% of RTP. A MSSV will be considered inoperable if it fails to open on demand. The LCO requires that five MSSVs be OPERABLE in compliance with the ASME Code, even though this is not a requirement of the DBA analysis. This is because operation with less than the

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(continued)

BASES (continued)

LCO (continued)

full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements), and adjustment to the Reactor Trip System setpoints. These limitations are addressed in Table 3.7.1-1 and Required Actions A.2.2.1 and A.2.2.2.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The MSSV setpoint and blowdown have been set to reduce the possibility of valve damage due to chattering. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Inspection and Testing Program.

The lift settings specified in Table 3.7.1-2 correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB.

APPLICABILITY

In MODE 1 above 43% RTP, the number of MSSVs per steam generator required to be OPERABLE must be as specified in Table 3.7.1-1. Below 43% RTP in MODE 1, 2, or 3, only two MSSVs per steam generator are required to be OPERABLE.

In MODE 4 or 5, there is no credible transient requiring the MSSVs.

The steam generators are not normally used for heat removal in MODE 5 or 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES. Should the steam generators be water-solid, however, it is prudent to have overpressure protection for them.

ACTIONS

<u>A.1</u>

With less than the required number of MSSVs per Table 3.7.1-1, verify that at least two required MSSVs per

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

steam generator are OPERABLE. Only two MSSVs per steam generator are required in MODE 2 or 3, and in MODE 1 below 43% RTP. Above 43% RTP, the number of MSSVs per steam generator required to be OPERABLE are governed by the applicable power level stated in Table 3.7.1-1.

This Action may be satisfied by examining logs or other information to determine whether the MSSVs are out of service for maintenance or other reasons. It does not mean that it is necessary to perform the SRs needed to demonstrate OPERABILITY of the MSSVs. The 4-hour Completion Time, which is the same as that for restoring a MSSV to OPERABLE status, is a reasonable amount of time to allow for examining information sources, such as maintenance logs, to determine whether two MSSVs are OPERABLE. The Completion Time takes into consideration the low probability of an event occurring during this period that would require activation of the MSSVs.

For this LCO, the Completion Times of Condition A have been provided with a Note to clarify that all MSSVs are treated as an entity with a single Completion Time (i.e., the Completion Times are on a condition basis).

A.2.1

If one or more MSSVs is inoperable, one alternative is to restore the required MSSVs to OPERABLE status per Table 3.7.1-1. Based on operating experience, the 4-hour Completion Time needed to restore a MSSV to OPERABLE status is reasonable, and takes into account the relative importance of maintaining the OPERABILITY of these valves, and the low probability of an event occurring during this period that would require activation of the MSSVs.

A.2.2.1 and A.2.2.2

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 applicable THERMAL POWER.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the

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(continued)

BASES (continued)

ACTIONS (continued)

remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. For example, if one MSSV is inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 20%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced by at least 20%. This is accomplished by reducing THERMAL POWER by at least 20%, which conservatively limits the energy transfer to all steam generators to approximately 80% of total capacity, consistent with the relief capacity of the most limiting steam generator. For example, if two MSSVs are inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 40%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced by at least 40%. This is accomplished by reducing THERMAL POWER by at least 40%, which conservatively limits the energy transfer to all steam generators to approximately 60% of total capacity, consistent with the relief capacity of the most limiting steam generator. Subsequently, if one MSSV is inoperable on a second and third steam generator, the relief capacity of these individual steam generators are reduced by approximately 20%, and energy transfers to these steam generators should be reduced by 20% by reducing THERMAL POWER to 80%. However, there are no further THERMAL POWER reductions required since THERMAL POWER was already reduced to 60% of total relief capacity for the first steam generator which in this case is the most limiting steam generator.

[For this facility, the values in Table 3.7.1-1 are calculated in the following manner for the pre-selected applicable power level and trip setpoint adjustments necessary to complete these Required Actions.]

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BASES (continued)

ACTIONS (continued)

For each steam generator, at a specified pressure, the fractional relief capacity (FRC) of each MSSV is determined as follows:

$$FRC = \frac{A}{B}$$

where:

A = the relief capacity of the MSSV; and

B = the total relief capacity of all the MSSVs of the steam generator

These FCRs are used to establish the power range neutron flux—high trip setpoints for operation with less than five MSSVs OPERABLE for each steam generator.

The reduced THERMAL POWER level and power range neutron flux—high trip setpoints specified in the LCO, prevent operation at power levels greater than the relief capacity of the remaining MSSVs. The reduced THERMAL POWER and trip setpoints are determined as follows:

$$RP = [1 - (N_1 \times FRC_1 + N_2 \times FRC_2 + \ldots\ldots + N_5 \times FRC_5)] \times 100\%$$

$$SP = RP \times T$$

where:

RP = Reduced THERMAL POWER for the most limiting steam generator expressed as a percent of RTP;

SP = Power Range Neutron Flux—High trip setpoints calculated for LCO 3.7.1 for operation at THERMAL POWER with less than five MSSVs OPERABLE;

$N_1, N_2, \ldots, N_5$ represent the status of the MSSV 1, 2,..., 5
= 0 if the MSSV is OPERABLE,
= 1 if the MSSV is inoperable;

$FRC_1, FRC_2, \ldots, FRC_5$ = the relief capacity of the MSSV 1, 2,..., 5 as defined above; and

T = ALLOWABLE VALUE for power range neutron flux—high trip setpoints as specified in LCO 3.3.1, "Reactor Trip System Instrumentation."

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BASES (continued)

ACTIONS (continued)

The 4-hour Completion Time for Required Action A.2.2.1 is consistent with A.2.1. An additional 4 hours is allowed to reduce the setpoints, in recognition of the difficulty of resetting all channels of this trip function within 4 hours. The Completion Time of 8 hours for Required Action A.2.2.2 is based on operating experience in resetting all channels of a protective function, and on the low probability of occurrence of a transient that could result in steam generator overpressure during this period.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR demonstrates the OPERABILITY of the MSSVs. Section XI, Article 3500 of the ASME Code (Ref. 4), requires that safety- and relief-valve tests be performed as required by ANSI/ASME OM-1-1987 (Ref. 5). Section 7.3.2.1 of Reference 5 requires the following tests for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Set pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The ANSI/ASME standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. Surveillances are specified in the Inservice Inspection and Testing Program, which encompasses Section XI of the ASME Code. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements.

SR 3.7.1.1 is modified by a Note that allows an exemption to SR 3.0.4. The MSSVs may be either bench tested or tested in-situ at hot conditions using an assist device to simulate lift pressure. The SR 3.0.4 exemption applies to those plants that have provisions for testing the MSSVs at hot conditions. It allows entry into and operation in MODE 3 for the performance of this surveillance. SR 3.0.4 is not applicable to this SR 3.7.1.1 provided testing is completed within 24 hours after reaching acceptable test conditions. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. [Unit Name] FSAR, Section [10.3.1], "[Main Steam System Description]."

2. ASME Boiler and Pressure Vessel Code, Section III, Article NC-7000, "Overpressure Protection," Class 2 Components.

3. [Unit Name] FSAR, Section [15.2], "[Decreased Heat Removal Events]."

4. ASME Boiler and Pressure Vessel Code, Section XI, Article IWV-3500, "Inservice Test: Category C Valves."

5. ANSI/ASME OM-1-1987, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices."

Table 3.7.1-1 (Page 1 of 1)

[VPHT] Setpoint
Versus OPERABLE MSSVs

MINIMUM NUMBER OF MSSVs PER SG REQUIRED OPERABLE	APPLICABLE POWER, % RTP	APPLICABLE TRIP SETPOINT, % RTP
[8	[	[
7		
6		
5		
4		
3		
2]	]	]

Table 3.7.1-2 (Page 1 of 1)

MSSV Lift Settings

VALVE NUMBER		LIFT SETTING
SG #1	SG #2	PSIG+3%

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break. MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, steam bypass system, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by either low steam generator pressure, or high containment pressure. The MSIVs fail closed on loss of control or actuation power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment (Ref. 2). It is also affected by the accident analysis of the SLB events presented in Reference 3. The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

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 lower powers, the steam generator inventory and temperature are at their maximum, maximizing

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIV contributes to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available, and with a loss-of-offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss-of-offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an MSIV to close.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. High energy line break inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture.

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

This LCO requires that the MSIV in each of the steam lines be OPERABLE.

The MSIVs are considered OPERABLE when their isolation times are within limits, and they close on an isolation actuation signal.

This LCO 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 or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

[For this facility, the support systems required OPERABLE to ensure the MSIVs OPERABILITY are as follows:]

[For this facility, those required support systems which, upon their failure, do not declare the MSIVs inoperable and their justification are as follows:]

(continued)

BASES (continued)

APPLICABILITY

The MSIVs must be OPERABLE whenever there is significant mass and energy in the RCS and steam generators. This ensures that in the event of a high energy line break, a single failure cannot result in the blowdown of more than one steam generator.

In MODE 1, 2, or 3, there is significant mass and energy; therefore, the MSIVs must be OPERABLE or closed. When the valves are closed, they are already performing their safety functions.

In MODE 4, normally most of the MSIVs are closed, and the steam generator energy is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

A Note has been added to provide clarification that the Completion Time is on a Condition basis; Conditions A and Conditions (C and D) Completion Times are independent.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs to the MSIV can be made with the plant hot. The 8-hour Completion Time is reasonable considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs.

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.

With more than one MSIV inoperable in MODE 1, the facility is in a condition outside the accident analysis; therefore, LCO 3.0.3 must be entered immediately.

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BASES (continued)

ACTIONS (continued)

B.1 and B.2

If the MSIV cannot be restored to OPERABLE status within 8 hours, the MSIV must be closed within the next 6 hours. Six hours is a reasonable time to complete the actions required to close the MSIV, which include performing a controlled plant shutdown to MODE 2. The Completion Time is based on plant operating experience related to the time required to reach MODE 2, with the MSIVs closed, in an orderly manner and without challenging plant systems.

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

With one MSIV inoperable in MODE 2 or 3 in one or more flow paths, MSIVs must be restored to OPERABLE status, or inoperable MSIVs must be closed within 8 hours. The remaining OPERABLE MSIV in each flow path will ensure that the flow path can be isolated, if an event were to occur that required isolation by these valves. The Completion Time of 8 hours takes into account the redundancy afforded by the OPERABLE MSIV, and the low probability of a Design Basis Accident that would require closure of the MSIVs occurring during this period.

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time but were closed, these inoperable valves must be verified to be continually closed on a periodic basis. This is necessary to ensure that the assumptions in the safety analyses remain valid. The 12-hour Completion Time is based on engineering judgment, and is considered reasonable in view of MSIV status indications available in the control room and other administrative controls which ensure that these valves will continue to be closed.

D.1 and D.2

With two MSIVs inoperable in MODE 2 or 3 in the same flow path for one or more flow paths, restore at least one MSIV to OPERABLE status in each affected flow path, or close at least one inoperable MSIV in each affected flow path within 1 hour. In this situation, the facility is in a condition outside the assumptions in the accident analyses. The

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ACTIONS (continued)

1-hour Completion Time provides a period of time to correct the problem, commensurate with the importance of bringing the facility within the assumptions of the accident analyses. This time period also ensures that the probability of an accident (requiring main steam line isolation) occurring during periods where two MSIVs are inoperable in the same flow path is minimal.

E.1 and E.2

The plant must be placed in a MODE in which the LCO does not apply if the MSIVs cannot be restored to OPERABLE status, or closed, within the associated Completion Time. This is done by placing the plant at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from MODE 2 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

The MSIV closure time is assumed in the accident and containment analyses. This surveillance is normally performed upon returning the plant to operation following a refueling outage. The MSIVs should not be tested at power, since even a part-stroke exercise increases the risk of a valve closure when the plant is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code Section XI (Ref. 4) requirements during operation in MODE 1 or 2.

The Frequency for this SR 3.7.2.1 is in accordance with the Inservice Inspection and Testing Program or 18 months, whichever is less. The 18-month surveillance Frequency to demonstrate valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This test is conducted in MODE 3 with the plant at operating temperature and pressure, as part of the ASME Code Section XI (Ref. 4) exercising requirements. SR 3.7.2.1 is modified by a Note that allows exemption to SR 3.0.4. SR 3.0.4 is

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

not applicable to SR 3.7.2.1 for entry into MODE 3 provided testing is completed within 24 hours after reaching acceptable test conditions. This allows a delay of testing in MODE 3, in order to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. [Unit Name] FSAR, Section [10.3], "[Main Steam System]."

2. [Unit Name] FSAR, Section [6.2], "[Containment Analysis]."

3. [Unit Name] FSAR, Section [15.1.5], "[Steam Line Break Analysis]."

4. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400, "Inservice Tests—Category A and B Valves."

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation and Regulation Valves (MFIVs and MFRVs) and Associated Bypass Valves

BASES

BACKGROUND

The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break. The safety related function of the MFRVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following a high energy line break. Closure of the MFIVs and associated bypass valves or MFRVs and associated bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs and associated bypass valves, or MFRVs and associated bypass valves, effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam or feedwater line breaks inside containment, and reducing the cooldown effects for steam line breaks (SLBs).

The MFIVs and associated bypass valves, or MFRVs and associated bypass valves, isolate the nonsafety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MFIV and its associated bypass valve, and one MFRV and its associated bypass valve are located on each MFW line, outside but close to containment. The MFIVs and MFRVs are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MFIV or MFRV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or feedwater line break (FWLB).

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BASES (continued)

BACKGROUND (continued)

The MFIVs and associated bypass valves, and MFRVs and associated bypass valves close on receipt of a T_{avg}—Low coincident with reactor trip (P-4) or steam generator water level—high-high signal. They may also be actuated manually. In addition to the MFIVs and associated bypass valves, and the MFRVs 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 and MFRVs is found in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs and MFRVs is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs and associated bypass valves, or MFRVs and associated bypass valves, may also be relied on to terminate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level—high-high signal or a main steam isolation signal on high steam generator level.

Failure of an MFIV, MFRV, or their associated bypass valves to close following an SLB or FWLB, can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs and MFRVs satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

Following an FWLB or main steam line break, this LCO ensures that the MFIVs and the MFRVs and their associated bypass valves will isolate main feedwater flow to the steam generators. These valves will also isolate the nonsafety-related portions from the safety-related portions of the system.

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BASES (continued)

LCO (continued)

This LCO requires that the MFIV and its associated bypass valve, and MFRV and its associated bypass valve, in each of the feedwater lines be OPERABLE. The MFIVs and MFRVs and their associated bypass valves are considered OPERABLE when their isolation times are within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. If a main steam isolation signal 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.

[For this facility, the support systems required OPERABLE to ensure the main feedwater valves OPERABILITY are as follows:]

[For this facility, those required support systems which, upon their failure, do not declare the main feedwater valves inoperable and their justification are as follows:]

APPLICABILITY

The MFIVs and MFRVs and their associated bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of a high energy line break, a single failure cannot result in the blowdown of more than one steam generator. In MODE 1, 2, or 3, the MFIVs and MFRVs and their associated bypass valves are required to be OPERABLE in order to limit the amount of available fluid that could be added to containment in case of a secondary system pipe break inside of containment. When the valves are closed, they are already performing their safety function.

In MODE 4, steam generator energy is low and the MFIVs and MFRVs and their associated bypass valves are normally closed since MFW is not required. In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water. Therefore, the MFIVs and MFRVs and their associated bypass valves are

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BASES (continued)

APPLICABILITY (continued)

not required for isolation of potential high energy secondary system pipe breaks in these MODES.

For this LCO, a Note has been added to provide clarification that Conditions A, B, and C are treated as an entity with a single Completion Time.

ACTIONS

A.1, A.2.1, A.2.2, B.1, B.2.1, and B.2.2

With one MFIV or its associated bypass valve, or one MFRV or its associated bypass valve in one or more flow paths inoperable, restore the affected valves to OPERABLE status, or close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function (i.e., to isolate the line).

The typical arrangement assumed in these Actions is one MFIV and one MFRV in series in the same feedwater line, and a bypass valve in parallel with either an MFIV or an MFRV, but not both. If a MFIV, MFRV, or bypass valve is inoperable and open under these conditions, then it is assumed that a second valve in the line will function automatically to isolate the line.

The 72-hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, and the low probability of an event that would require isolation of the main feedwater flow paths occurring during this period.

For inoperable MFIVs, MFRVs and their associated bypass valves that cannot be restored to OPERABLE status within the specified Completion Time but were closed or isolated, these inoperable valves must be verified to be continually closed or isolated on a periodic basis. This is necessary to ensure that the assumptions in the safety analyses remain valid. The 12-hour Completion Time is based on engineering judgment and is considered reasonable in view of valve status indications available in the control room and other administrative controls that will ensure that these valves will continue to be in the closed or isolated position.

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BASES (continued)

ACTIONS (continued)

C.1 and C.2

If more than one valve in the same flow path cannot be restored to OPERABLE status, then there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure is likely to be a precursor 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, closed, or the affected flow path isolated within 8 hours. This action returns the system to the situation in which at least one valve in each flow path is performing the required safety function. The 8-hour Completion Time is a reasonable amount of time to complete the actions required to close the MFIV or MFRV, which includes performing a controlled plant shutdown to MODE 2. The Completion Time is based on plant operating experience related to the time required to reach MODE 2 with the MSIVs closed in an orderly manner and without challenging plant systems.

D.1 and D.2

If the MFIVs and MFRVs and their associated bypass valves cannot be restored to OPERABLE status, closed or isolated within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

The MFIV and MFRV closure times are assumed in the accident and containment analyses. This surveillance is normally performed upon returning the plant to operation following a refueling outage. These valves should not be tested at

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	power since even a part-stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not tested at power, they are exempt from the ASME Section XI requirements during operation in MODES 1 and 2. The Frequency for this SR 3.7.3.1 is in accordance with the Inservice Inspection and Testing Program or 18 months, whichever is less. The 18-month surveillance Frequency to demonstrate valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the surveillances when performed on the 18-month Frequency. SR 3.7.3.1 is modified by a Note which allows exemption to SR 3.0.4. SR 3.0.4 is not applicable to SR 3.7.3.1 for entry into MODE 3 provided testing is completed within 24 hours after reaching acceptable test conditions. This allows delaying testing in MODE 3 in order to establish conditions consistent with the conditions under which the acceptance criterion was generated.
REFERENCES	1. [Unit Name] FSAR, Section [10.4.7], "[Condensate and Feedwater System]."

B 3.7 PLANT SYSTEMS

B 3.7.4 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND	The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through separate and independent suction lines from the condensate storage tank (CST) (LCO 3.7.5) and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.11). If the main condenser is available, steam may be released via the steam bypass valves and recirculated to the CST.

The AFW System consists of [two] motor-driven AFW pumps and [one] steam turbine-driven pump configured into [three] trains. Each motor-driven pump provides [100]% of AFW flow capacity, and the turbine-driven pump provides [200]% of the required capacity to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor-driven AFW pump is powered from an independent Class 1E power supply and feeds [two] steam generators, although each pump has the capability to be realigned from the control room to feed other steam generators. The steam turbine-driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine-driven AFW pump.

The AFW System supplies feedwater to the steam generators during normal plant startup, shutdown, and hot standby conditions.

The turbine-driven AFW pump supplies a common header capable of feeding all steam generators with dc powered control valves actuated to the appropriate steam generator by the

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BASES (continued)

BACKGROUND (continued)

Engineered Safety Feature Actuation System (ESFAS). One pump at full flow is sufficient to remove decay heat and cool the plant to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the plant to RHR entry conditions, with steam released through the ADVs.

The AFW System actuates automatically on steam generator water level—low-low by the ESFAS (LCO 3.3.2). The system also actuates on loss-of-offsite power, safety injection, and trip of all MFW pumps.

The AFW System is discussed in Reference 1.

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW 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 AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW 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 AFW System are:

a. Feedwater system pipe rupture; and

b. Loss of normal feedwater.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small-break loss-of-coolant accident (LOCA).

The AFW System design is such that it can perform its function following a feedwater line break between the main feedwater isolation valves and containment, combined with a loss-of-offsite power following turbine trip, and a single active failure of the steam turbine-driven AFW pump. In such a case, the ESFAS logic may not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly. One motor-driven AFW pump would deliver to the broken MFW header at the pump runout flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine-driven pump and associated DC power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of AC power. DC power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement.

LCO

This LCO provides assurance that the AFW 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 AFW pumps in [three] diverse trains 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 of the pumps from independent emergency buses. The third AFW pump is powered by a different means, a steam-driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs.

The AFW System is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the

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BASES (continued)

LCO (continued)

steam generators are OPERABLE. This requires that the two motor-driven AFW pumps be OPERABLE in [two] diverse paths, each supplying AFW to separate steam generators. The turbine-driven AFW pump shall be OPERABLE with redundant steam supplies from each of [two] main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE.

The LCO is modified by a Note requiring that only one motor-driven pump be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine-driven AFW pump.

[For this facility, each OPERABLE AFW System train consists of the following:]

[For this facility, support systems required to be OPERABLE to ensure the AFW System OPERABILITY are as follows:]

[For this facility, those required support systems which, upon their failure, do not declare the AFW System inoperable and their justification are as follows:]

APPLICABILITY

In MODE 1, 2, or 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the main feedwater is lost. In addition, the AFW System is required to supply enough makeup water to replace steam generator secondary inventory that is lost as the unit cools to MODE 4 conditions.

In MODE 4, without any RHR loops OPERABLE or in operation, the AFW System is required to be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

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BASES (continued)

APPLICABILITY (continued)

For this LCO, a Note has been added to provide clarification that all components of the AFW trains are treated as an entity with a single Completion Time.

ACTIONS

A.1

If one of the two steam supplies to the turbine-driven AFW train is inoperable, action must be taken to restore the steam supply to OPERABLE status. The 7-day Completion Time is justified based on:

a. The redundant OPERABLE steam supply to the turbine-driven AFW pump;

b. The availability of redundant OPERABLE motor-driven AFW pumps; and

c. The low probability of an event occurring that requires the inoperable steam supply to the turbine-driven AFW pump.

B.1

If one of the required AFW trains (pump or flow path) is inoperable, action must be taken to restore the train to OPERABLE status. This Condition includes the loss of two steam supply lines to the turbine-driven AFW pump. The 72-hour Completion Time was chosen because of the redundant capabilities afforded by the AFW System, reasonable time for repairs, and the low probability of a DBA occurring during this period.

C.1 and C.2

When either Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours, and in MODE 4 (except as indicated in a Note applicable to Required Action C.2) within 18 hours.

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BASES (continued)

ACTIONS (continued)

Required Action C.2 has been modified by a Note intended to restrict entry into MODE 4 when no RHR loops are OPERABLE and in operation. The Note also is intended to convey the suspension of further action to reach MODE 4 if, while in Required Action C.2, all RHR loops become inoperable or not in operation. The allowed Completion Times are reasonable, based on operating experience, to reach the conditions from full power in an orderly manner and without challenging plant systems.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor-driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the plant may continue to cool down and initiate RHR, if prudent.

D.1

If all three AFW trains are inoperable, the plant is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with non-safety-related equipment. In such a condition, the plant 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 AFW train to OPERABLE status.

Required Action D.1 is modified by a Note to suspend all required MODE changes or power reductions until at least one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the AFW System water and steam flow paths, provides assurance that the proper flow paths will exist for AFW operation. This SR 3.7.4.1 does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.1 also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31-day Frequency of this SR 3.7.4.1 was derived from Inservice Inspection and Testing Program requirements for performing valve testing at least once every 92 days. This increased surveillance Frequency is further justified in view of the procedural controls governing valve operation, and as a means of providing added assurance of correct valve positions.

SR 3.7.4.2

This SR demonstrates that the AFW pumps develop sufficient discharge pressure to deliver the required flow at the full open pressure of the MSSVs. Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. Periodically comparing the reference differential pressure developed at this reduced flow, detects trends that might be indicative of incipient failures. Performing ASME Section XI (Ref. 2) inservice testing (only required at 3-month intervals) satisfies this requirement, as per the Inservice Inspection and Testing Program.

A 31-day Frequency on a STAGGERED TEST BASIS results in testing each pump once per 3 months, as required by the ASME Code.

Provisions of SR 3.7.4.2 are modified by a Note that allows an exception to SR 3.0.4. SR 3.0.4 are not applicable for entry into MODE 3 for purposes of testing the turbine-driven AFW pump due to an insufficient amount of steam in MODE 4, 5, or 6 to perform a valid test.

SR 3.7.4.3

This SR ensures that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. The 18-month Frequency was developed considering the plant conditions needed to perform SR 3.7.4.3, and the potential REQUIREMENTS for unplanned plant transients if SR 3.7.4.3 is 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 every 92 days as part of the Safety Injection Actuation System's functional test, except for the subgroup relay that actuates the system, which cannot be tested during normal plant operations.

SR 3.7.4.4

This SR ensures that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The actuation logic is tested as part of the Safety Injection Actuation System's functional test every 92 days, except for the subgroup relay that actuates the system, which cannot be tested during normal plant operation. The 18-month Frequency justification is the same as that for SR 3.7.4.3.

This SR is modified by a Note that exempts the requirements of SR 3.0.4 for entry into MODE 3 for purposes of testing the turbine-driven AFW pump, due to an insufficient amount of steam in MODE 4, 5, or 6 to perform a valid test. SR 3.0.4 is not applicable to SR 3.7.4.4 for entry into MODE 3, provided testing is completed within 24 hours after reaching acceptable test conditions.

SR 3.7.4.5

This SR ensures that the AFW is properly aligned by demonstrating the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is based on engineering judgment, and is considered adequate in view cf other administrative controls

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

that ensure that flow paths remain OPERABLE. For added assurance, the OPERABILITY of the flow paths is verified following extended outages to determine that there is no misalignment of valves. SR 3.7.4.5 ensures that the flow path from the CST to the steam generators is properly aligned. (SR 3.7.4.5 is not required by those plants REQUIREMENTS that use AFW for normal startups and shutdowns.)

REFERENCES

1. [Unit Name] FSAR, Section [10.4.9], "[Auxiliary Feedwater System]."

2. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400, "Inservice Tests—Category A and B Valves."

B 3.7 PLANT SYSTEMS

B 3.7.5 Condensate Storage Tank (CST)

BASES

BACKGROUND	The CST provides a safety-grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.4). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric dump valves. The AFW pumps operate with a continuous recirculation to the CST. When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the non-safety-grade path of the steam bypass valves. The condensed steam is returned to the CST by the condensate transfer 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, including 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 alternate sources. [For this facility, the alternate sources of feedwater and their safety-grade classifications are as follows:] A description of the CST is found in Reference 1.
APPLICABLE SAFETY ANALYSES	The CST provides cooling water to remove decay heat and to cool down the plant following all events in the accident analysis (Refs. 2 and 3). 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 residual heat removal (RHR) entry conditions at the design cooldown rate. (continued)

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The limiting event for the condensate volume is the large feedwater line break coincident with a loss-of-offsite power. Single failures that also affect this event include:

a. Failure of the diesel generator powering the motor-driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump's turbine); and

b. Failure of the steam-driven AFW pump (requiring a longer time for cooldown using only one motor-driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

A nonlimiting event considered in CST inventory determinations is a break in either the main feedwater or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, since the Emergency Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location. This loss-of-condensate inventory is partially compensated for by the retention of steam generator inventory.

The CST satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for [30 minutes] following a reactor trip from 102% RATED THERMAL POWER, and then to cooldown the RCS to RHR entry conditions, assuming a coincident loss-of-offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam-driven AFW pump's turbines, or before isolating AFW to a broken line.

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BASES (continued)

LCO (continued)

The level required is equivalent to a usable volume of [110,000] gallons, which is based on holding the plant in MODE 3 for 2 hours, followed by a cooldown to RHR entry conditions at [75]°F/hour. This basis is established in Reference 4 and exceeds the volume required by the accident analysis.

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

[For this facility, the following support systems are required to be OPERABLE to ensure CST OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring CST inoperable and their justification are as follows:]

APPLICABILITY

In MODE 1, 2, 3, or 4, the applicability of the CST is consistent with AFW System applicability (LCO 3.7.4), since the CST directly supports the AFW System.

In MODE 5 or 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1

If the CST is unable to supply the required volume of cooling water to the AFW pumps, it must be restored to OPERABLE status. Four hours allow time to restore the required volume in the CST from the condenser, or backup supply, and is a reasonable time to limit the risk from accidents requiring the plant to cool down.

In the event that the required CST water level channels are determined to be inoperable, the CST level is considered to be not within limits; Required Action A.1 and Required Action A.2.2 apply to restore such equipment to OPERABLE status.

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(continued)

BASES (continued)

ACTIONS (continued)

A.2.1 and A.2.2

As an alternative to shutting down the unit, the OPERABILITY of the backup supply may be verified before 4 hours expires. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the AFW pumps are OPERABLE, and that the backup supply has the required volume of water available. The CST must still be returned to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. Based on operating experience, the 4-hour Completion Time is a reasonable amount of time to verify the OPERABILITY of the backup wa supply. The 7-day Completion Time is reasonable in view the fact that an OPERABLE backup water supply is avail and the low probability of an event requiring CST oc during this period.

[For this facility, an OPERABLE backup water supply co of the following:]

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO requirements do not apply. This is done by placing the plant in at least MODE 3 within 6 hours, and in MODE 4 (except as indicated in a Note applicable to Required Action B.2) within 18 hours. Required Action B.2 has been modified by a Note intended to restrict entry into MODE 4 without any RHR loops OPERABLE and in operation. The Note also is intended to convey the suspension of further action to reach MODE 4 if, while in Required Action B.2, all RHR loops became inoperable or not in operation. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

SR 3.7.5.1 verifies that the CST contains the required volume of cooling water. (The required CST volume may be

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

single value or a function of RCS conditions.) The 12-hour Frequency of this SR was developed based on consideration of operating experience and operator awareness of plant evolutions that may affect the CST inventory between checks. Also, the 12-hour Frequency is considered adequate in view of other indications, including alarms available in the control room to alert the operator of abnormal deviations in CST level.

REFERENCES

1. [Unit Name] FSAR, Section [9.2.6], "[Condensate Storage and Transfer System]."
2. [Unit Name] FSAR, Section [6], "[Title]."
3. [Unit Name] FSAR, Section [15], "[Accident Analysis]."
4. NRC Standard Review Plan, Branch Technical Position RSB 5-1, "Design Requirements for the Residual Heat Removal System."

B 3.7 PLANT SYSTEMS

B 3.7.6 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System. Under steady-state conditions, the activity is primarily iodines with relatively short half-lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission-product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environs 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) of primary coolant at the limit of 1.0 μCi/gram (LCO 3.4.16). 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., less than 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 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.

Operating a plant 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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BASES (continued)

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB) (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.1 μCi/g DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the plant EAB limits of 10 CFR 100 for whole-body and thyroid dose rates.

With the loss-of-offsite power, the remaining steam generators are available for core-decay-heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure has decreased sufficiently for the Residual Heat Removal 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 satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

As indicated in the applicable safety analyses, the specific activity limit in the secondary coolant system of ≤ 0.1 μCi/g DOSE EQUIVALENT I-131 maintains the radiological consequences of a Design Basis Accident (DBA) to a small fraction of 10 CFR 100 (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are

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BASES (continued)

LCO (continued)	exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.
APPLICABILITY	In MODE 1, 2, 3 or 4, the limits on secondary specific activity apply whenever using the steam generators for RCS heat removal. This is a potential time for secondary steam releases to the atmosphere, carrying with the steam a portion of the activity in the steam generators. In MODE 5 or 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary-to-secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.
ACTIONS	A.1 and A.2 DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits with in the associated Completion Time, the plant must be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.6.1 This SR 3.7.6.1 ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post-accident releases. It also serves to identify and trend any unusual

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31-day frequency takes into consideration the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. [Unit Name] FSAR, Section [15], "[Accident Analysis]."

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND	The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a transient or accident. 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. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection (SI) signal, and all nonessential components are isolated. [For this facility, the CCW System consists of the following:] Additional information on the design and operation of the system, along with a list of the components served, is presented in Reference 1. The principal safety related function of the CCW System function is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. This may be during a normal or post-accident cooldown and shutdown.
APPLICABLE SAFETY ANALYSES	The design basis of the CCW System is for one CCW train to remove the post loss-of-coolant accident (LOCA) heat load from the containment sump during the recirculation phase, with a maximum CCW temperature of [120]°F (Ref. 2). The Emergency Core Cooling System (ECCS) LOCA, and containment OPERABILITY LOCA each model the maximum and minimum performance of the CCW System, respectively. The normal

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

temperature of the CCW is [80]°F and, during plant cooldown to MODE 5 (T_{cold} < [200]°F), a maximum temperature of [95]°F is assumed. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the SI pumps.

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 plant from RHR entry conditions (T_{cold} < [350]°F), to MODE 5 (T_{cold} < [200]°F) during normal and post-accident operations. The time required to cool from [350]°F to [200]°F is a function of the number of CCW and RHR trains operating. One train is sufficient to remove decay heat during subsequent operations with T_{cold} < [200]°F. This assumes a maximum service water temperature of [95]°F occurring simultaneously with the maximum heat loads on the system

The CCW System satisfies Criterion 3 of the NRC Interim 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 does not depend on the other. In the event of a Design Basis Accident (DBA), one train of CCW is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two trains of CCW must be OPERABLE. At least one train will operate assuming the worst-case single active failure occurs coincident with a loss-of-offsite power.

A train is considered OPERABLE when:

a. Its pump and associated surge tank are OPERABLE; and

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BASES (continued)

LCO (continued)

b. The associated piping, valves, heat exchanger, instrumentation, and controls on the safety-related flow path are OPERABLE.

The isolation of CCW from other components or systems not required for safety may render those components or system inoperable, but does not affect the OPERABILITY of the CCW System.

[For this facility, the following support systems are required to be OPERABLE to ensure CCWS OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not declare the CCWS inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of the CCWS and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

In MODE 1, 2, 3, or 4, the CCW System is a normally operating system, which must be prepared to perform its post-accident safety functions, primarily RCS heat removal, which is achieved by cooling the RHR heat exchanger.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

If only one CCW train is OPERABLE, the inoperable CCW train must be restored to 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 was developed taking into account the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

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BASES (continued)

ACTIONS (continued)

B.1

With one CCW train inoperable, verify that the Required Actions have been initiated for those supported systems declared inoperable by the support CCW train within a Completion Time of [ ] hours.

The [ ]-hour Completion Time is defined as the most limiting of all the Required Actions for all the supported systems that needed to be declared inoperable upon the failure of one or more support features specified under Condition B.

Required Action B.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of CCW trains have been initiated by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition B of this LCO.]

[For this facility, the identified supported systems' Required Actions are as follows:]

C.1

With one CCW train inoperable, AND one or more required support or supported features inoperable associated with the other redundant CCW train; enter Required Actions of Condition D. Condition C is indicative of loss of CCW System functional capability.

D.1, D.2, D.3, and D.4

If the CCW train cannot be restored to OPERABLE status within the associated Completion Time, or two CCW trains are inoperable, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

If both trains are inoperable, immediate action must be taken to restore at least one train to OPERABLE status. In this case, there is no heat sink for the RHR System, thus one CCW train must be restored to OPERABLE status and the plant should be maintained in MODE 4, where decay heat can be removed by the steam generators.

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BASES (continued)

ACTIONS (continued)

With both trains inoperable, flexibility (and abnormal operating procedures) are left to the operator to manage the situation. This allows remaining in MODE 4 with an alternate means of heat removal. This action allows total loss of function without entry into MODE 5 as required by LCO 3.0.3, which may not be possible with two CCW trains inoperable. When a CCW train is OPERABLE. the plant should then be placed in MODE 5. In this case, LCO 3.0.3 is not applicable, since the plant cannot be brought to MODE 5 without at least one train of CCW OPERABLE. Adequate heat removal can be maintained using the steam generators and natural circulation.

The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves 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 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 of this SR was derived from Inservice Inspection Testing Program requirements for performing valve testing at least once every 92 days. The Frequency is further justified in view of the procedural controls governing valve operation, and as a means of providing added assurance of correct valve positions.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.2

SR 3.7.7.2 demonstrates proper automatic operation of the CCW valves. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 18-month Frequency was developed considering the plant conditions needed to perform the surveillance and the potential for unnecessary plant transients if the surveillance is performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.7.7.3

SR 3.7.7.3 demonstrates proper automatic operation of the CCW pumps. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 18-month Frequency was developed considering the plant conditions needed to perform the surveillance and the potential for unnecessary plant transients if the surveillance is performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. [Unit Name] FSAR, Section [9.2.2], "[Component Cooling Water System]."

2. [Unit Name] FSAR, Section [6.2], "[Containment Analysis]."

B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety-related components during a transient or Design Basis Accident (DBA). During normal operation, and a normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related portion is covered by this LCO.

A typical SWS consists of two separate, 100%-capacity safety-related, cooling water trains. Each train consists of two 100%-capacity pumps, one component cooling water (CCW) heat exchanger, piping, valving, instrumentation, and two cyclone separators. The pumps and valves are remote and manually aligned, except in the unlikely event of a loss-of-coolant accident (LOCA). The pumps aligned to the critical loops are automatically started upon receipt of a safety injection (SI) signal, and all essential valves are aligned to their post-accident positions. The SWS also provides emergency makeup to the Spent Fuel Pool and CCW System [and is the backup water supply to the Auxiliary Feedwater System]. [For this facility, the SWS consists of the following:]

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in Reference 1. The principal safety related function of the SWS is the removal of decay heat from the reactor via the [CCW System].

APPLICABLE SAFETY ANALYSES

The design basis of the SWS is for one SWS train, in conjunction with the CCW System and a 100% capacity Containment Cooling System, to remove core decay heat following a design basis LOCA (Ref. 2). This prevents the Containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the SI pumps. The SWS is designed to perform its function with a single

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

failure of any active component, assuming the loss-of-offsite power.

The SWS, in conjunction with the CCW System, also cools the plant from residual heat removal (Ref. 3) entry conditions to MODE 5 during normal and post-accident operations. The time required for this evolution is a function of the number of CCW and RHR 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 [95]°F occurring simultaneously with maximum heat loads on the system.

The SWS satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

Two SWS trains provide the required redundancy to ensure that the system functions to remove post-accident heat loads, assuming that the worst-case, single active failure occurs coincident with the loss-of-offsite power.

A train is considered OPERABLE during MODE 1, 2, 3, or 4 when:

1. Its pump is OPERABLE; and
2. The associated piping, valves, heat exchanger, instrumentation, and cyclone separator on the safety related flow path are OPERABLE.

The isolation of the SWS to other components or systems not required for safety may render these components or systems inoperable, but does not affect the OPERABILITY of the SWS.

[For this facility during MODE 1, 2, 3 or 4, the following support systems are required to be OPERABLE to ensure SWS OPERABILITY:]

[For this facility during MODE 1, 2, 3 or 4, those required support systems which, upon their failure, do not require declaring the SWS inoperable and their justification are as follows:]

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BASES (continued)

LCO (continued)

[For this facility during MODE 1, 2, 3 or 4, the supported systems impacted by the inoperability of a SWS and the justification of whether or not each supported system is declared inoperable are as follows:]

[For this facility during MODE 5 or 6, the supported systems impacted by the inoperability of a SWS and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

In MODE 1, 2, 3, or 4, the SWS is a normally operating system that must be prepared to perform its post-accident safety functions, primarily RCS heat removal, by cooling the CCW System and, thus, the RHR System.

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

ACTIONS

A.1

If one SWS train is inoperable, it must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SWS train is adequate to perform the heat removal function. The 72-hour Completion Time was developed based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1

With one SWS train inoperable, verify that the Required Actions have been initiated for those supported systems declared inoperable by the support SWS train within a Completion Time of [ ] hours.

The [ ]-hour Completion Time is defined as the most limiting of all the Required Actions for all the supported systems that needed to be declared inoperable upon the failure of one or more support features specified under Condition B.

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BASES (continued)

ACTIONS (continued)

Required Action B.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of SWS trains have been initiated by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition B of this LCO.]

[For this facility, the identified support systems' Required Actions are as follows:]

C.1

With one SWS train inoperable, and one or more required support or supported features inoperable associated with the other redundant SWS train, enter Required Actions of Condition D. Condition C is indicative of a loss-of-SWS functional capability.

D.1, D.2, D.3, and D.4

If the SWS train cannot be restored to OPERABLE status within the associated Completion Time, or two SWS trains are inoperable, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

If both trains are inoperable, action must be taken to restore at least one train to OPERABLE status. In this case, there is no heat sink for the RHR System; thus, one SWS train must be restored to OPERABLE status immediately. The plant should be maintained in MODE 4 until one SWS train is restored to OPERABLE status.

When a SWS train is OPERABLE, the plant should be placed in MODE 5. This allows total loss of function without entry into LCO 3.0.3, because entry into MODE 5, as required by LCO 3.0.3, may not be desirable with two SWS trains inoperable.

The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

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BASES (continued)

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. SR 3.7.8.1 does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to being locked, sealed, or secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31-day Frequency for this surveillance was derived from Inservice Inspection and Testing Program requirements for performing valve testing at least once every 92 days. The Frequency was further justified in view of the procedural controls governing valve operation, and as a means of providing added assurance of correct valve position.

SR 3.7.8.2

SR 3.7.8.2 demonstrates proper automatic operation of the SWS valves. The SWS is a normally operating system that cannot be fully actuated as part of normal testing. The 18-month Frequency was developed because this surveillance can only be prudently performed during a plant outage. This is due to the plant conditions needed to perform the surveillance and the potential for unnecessary plant transients if the surveillance is performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.7.8.3

SR 3.7.8.3 demonstrates proper automatic operation of the SWS pumps. The SWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 18-month Frequency was developed because this surveillance can only be prudently performed

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

during a plant outage. This is due to the plant conditions needed to perform the surveillance, and the potential for unnecessary plant transients if the surveillance is performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. [Unit Name] FSAR, Section [9.2.1], "[Service Water System]."

2. [Unit Name] FSAR, Section [6.2], "[Containment Analysis]."

3. [Unit Name] FSAR, Section [5.4.7], "[Residual Heat Removal]."

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Service Water System (SWS) and the Component Cooling Water (CCW) System.

[The UHS consists of the lake, SWS pumps, CCW heat exchangers, cooling towers, fans, and associated piping, valves, and instrumentation.]

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 (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 safety functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

A variety of complexes is used to meet the requirements for an 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 have been 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 are not exceeded. Basins of cooling towers generally include less than a 30-day supply of water, typically 7 days or less. Assurance of a 30-day supply is then dependent on other source(s) and 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

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BASES (continued)

BACKGROUND (continued)

Feature (ESF) (e.g., single failure considerations), and multiple makeup water sources may be required.

It follows that the many variations in the UHS configurations will result in many plant-to-plant variations in OPERABILITY determinations and in SRs. The Actions and SRs are illustrative of a cooling tower UHS without a makeup requirement. [Development of UHS Technical Specifications for plants without cooling towers and makeup systems may require Actions and Surveillances for components in addition to cooling tower fans (e.g. makeup pumps and isolation valves).]

[For plants without cooling towers, additional Actions and SRs may be necessary (e.g., a second source or use of spray ponds).]

Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1. [For this facility, the UHS consists of the following:]

If the UHS does not meet its design limits of water temperature, water level, or number of OPERABLE cooling tower fans, the UHS may not have sufficient capacity to bring the plant to a safe, controlled shutdown during a Design Basis Accident (DBA) from full power, but may be able to support plant operation at a reduced power level.

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences (AOOs) in which the plant is cooled down and placed on residual heat removal (RHR) operation. [For plants that use it as the normal heat sink for condenser cooling via the Circulating Water System, plant operation at full power is its maximum heat load.] Its maximum post-accident heat load occurs 20 minutes after a design basis loss-of-coolant accident (LOCA). Near this time, the plant switches from injection to recirculation and the Containment Cooling Systems and RHR are required to remove the core decay heat.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The operating limits are based on conservative heat transfer analyses for the worst-case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst-expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst-case single active failure (e.g., single failure of a man-made 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 Interim Policy Statement.

LCO

The UHS 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 plant operation.

[For this facility, an OPERABLE UHS consists of the following:]

[For this facility, the following support systems are required OPERABLE to ensure the UHS OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not declare the UHS inoperable and their justification are as follows:]

[For this facility, the main systems supported by the UHS, and the justification for not declaring the main systems inoperable upon failure of the UHS are as follows:]

APPLICABILITY

In MODE 1, 2, 3, or 4, the UHS is required to be OPERABLE to ensure sufficient cooling capacity and must be prepared to perform its post accident safety functions. An example is Reactor Coolant System heat removal for core decay heat.

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1 and A.2

Required Action A.1 ensures that the required cooling capacity will be available in the event of a DBA.

This Action may be satisfied by examining logs or other information to determine whether the cooling tower fans may be out of service for maintenance or other reasons. It does not mean that it is necessary to perform the SRs needed to demonstrate OPERABILITY of the fan. If there is not one cooling tower fan per cooling tower OPERABLE, Condition D must be entered immediately.

For Action A.2, if one cooling tower fan per cooling tower is inoperable, the inoperable cooling tower fans must be restored to OPERABLE status within 7-days before action must be taken to reduce power. The specified Completion Time is consistent with other LCOs for loss of one-half of a 200%-capacity train of an ESF System.

The 7-day Completion Time is based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable, the number of available systems, and the time required to reasonably complete the Required Action.

For this LCO, the Completion Times of Condition A have been provided with a Note to clarify that all UHS cooling tower fans are treated as an entity with a single Completion Time (i.e., the Completion Times are on a Condition basis).

B.1

With the UHS inoperable as established by Condition D, verify that the Required Actions have been initiated for those supported systems declared inoperable by the support UHS within a Completion Time of [ ] hours.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

The [ ]-hour Completion Time is defined as the most limiting of all the Required Actions for all the supported systems that needed to be declared inoperable upon the failure of one or more support features specified under Condition B.

Required Action B.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of UHS have been initiated by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition B of this LCO.]

[For this facility, the identified supported systems' Required Actions are as follows:]

C.1

With one or more cooling tower fans inoperable, and one or more required support or supported features associated with the other redundant cooling tower fan inoperable; a loss of function capability results, and LCO 3.0.3 must be entered immediately. However, if the support or supported features' LCOs take into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

D.1 and D.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 plant must be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 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 MODES from full power in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

SR 3.7.9.1 ensures that adequate long-term (30-day) cooling can be maintained. The specified level ensures that eno h NPSH is available to operate the SWS pumps. The 24-hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.9.2

SR 3.7.9.2 verifies that the SWS can cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a DBA. The 24-hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.9.3

Operating each cooling tower fan for ≥ 15 minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31-day Frequency was developed in consideration of the 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. It has also been shown to be acceptable through operating experience.

REFERENCES

1. [Unit Name] FSAR, Section [9.2.5], "[Ultimate Heat Sink]."

2. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 2, January 1976.

B 3.7 PLANT SYSTEMS

B 3.7.10 Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the fuel storage pool meets the assumptions of iodine-decontamination factors following a fuel-handling accident. The specified water level shields and minimizes the general area dose when the storage racks are at their maximum capacity. The water also provides shielding during the movement of spent fuel. If normal cooling is lost, the water provides about a 12-hour heat sink before boiling occurs. A general description of the fuel storage pool design is given in Reference 1. A description of the Spent Fuel Pool Cooling and Cleanup System is given in Reference 2. The assumptions of the fuel-handling accident are given in Reference 3.
APPLICABLE SAFETY ANALYSES	The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2-hour thyroid dose per person at the exclusion area boundary (EAB) is a small fraction of the 10 CFR 100 (Ref. 5) limits. The assumption of Regulatory Guide 1.25, preserved by this LCO, is that there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel-handling accident. With 23 ft of water, the assumptions of Regulatory Guide 1.25 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be less than 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first [few] rods fail from a hypothetical maximum drop. The fuel storage pool water level satisfies Criterion 2 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

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.

[For this facility, an OPERABLE fuel storage pool constitutes the following:]

[For this facility, the following support systems are required OPERABLE to ensure fuel storage pool water level OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not declare the fuel storage pool water level inoperable and their justification are as follows:]

APPLICABILITY

This LCO applies whenever irradiated fuel is in the spent fuel storage pool, because the potential for a release of fission products exists.

ACTIONS

A.1

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool level is lower than the required level, the movement of spent fuel shall be brought to a halt in a safe position. This effectively precludes a fuel-handling accident from occurring. Plant procedures control the movement of loads over the spent fuel in all cases.

In the event that the required fuel storage pool water level channels are determined inoperable, the fuel storage pool water cover is considered to be not within limits; Required Action A.1 and Required Action A.2 apply to restore the instrumentation channels to OPERABLE status.

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(continued)

BASES (continued)

ACTIONS (continued)

A.2

Action to restore the water level should commence immediately and be carried through to completion.

Required Action A.1 and Required Action A.2 are modified by a Note that allows an exemption to LCO 3.0.3 and LCO 3.0.4. Both LCOs are not applicable, since events in the fuel storage pool are not affected by either MODE level or facility operations.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

SR 3.7.10.1 verifies that sufficient 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 in view of the fact that the volume in the pool is normally stable. Water level changes are controlled by plant procedures and have been proven to be acceptable through operating experience.

During refueling operations, the level in the fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily under SR 3.9.6.1.

REFERENCES

1. [Unit Name] FSAR, Section [9.1.2], "[Spent Fuel Storage]."

2. [Unit Name] FSAR, Section [9.1.3], "[Spent Fuel Pool Cooling and Cleanup System]."

3. [Unit Name] FSAR, Section [15.7.4], "[Fuel Handling Accident]."

4. Regulatory Guide 1.25 [Rev. 00], "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

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BASES (continued)

REFERENCES (continued)	5.	Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

B 3.7 PLANT SYSTEMS

B 3.7.11 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a method for cooling the plant to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Bypass System to the condenser not be available (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST). The ADVs 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 Steam Dump System.

One ADV line for each of the [four] steam generators is provided. Each ADV line consists of one ADV and an associated block valve.

The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are typically provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in that the normal instrument air supply, automatically supplies nitrogen to operate the ADVs. The nitrogen supply is sized to provide the sufficient pressurized gas to operate the ADVs for the time required for Reactor Coolant System (RCS) cooldown to RHR entry conditions.

A description of the ADVs is found in Reference 1. The ADVs are OPERABLE with only a dc power source available. In addition, handwheels are provided for local manual operation. [For this facility, the ADVs and their support systems consist of the following:]

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the plant to RHR entry conditions. The design rate of [75]°F/hour is applicable for two steam

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(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

generators, each with one ADV. This rate is adequate to cool the plant to RHR entry conditions with only one steam generator and one ADV, utilizing the cooling water supply available in the CST.

In the accident analysis presented in the FSAR, the ADVs are assumed to be used by the operator to cool down the plant to RHR entry conditions for accidents accompanied by a loss-of-offsite power. Prior to the operator's actions to cool down the plant, the ADVs and main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (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. After the break flow is terminated, the operator would then continue the cooldown to RHR conditions, similar to the cooldown for other events. The operator is assumed to use only the ADVs on the nonruptured steam generators to perform the limited cooldown required to terminate the break flow and, subsequently, to cool down the plant to RHR entry conditions. The time required to terminate the primary-to-secondary break flow for a SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for the ADVs. The number of ADVs required to be OPERABLE to satisfy the SGTR accident analysis requirements depends upon the number of plant loops and consideration of any single failure assumptions regarding the failure of one ADV to open on demand.

The ADVs are equipped with block valves in the event an ADV spuriously fails to open or fails to close during use.

The ADVs satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

One ADV line is required from each of [four] steam generators to ensure that at least [two] ADV lines are available to conduct a plant cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second ADV line on an

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(continued)

BASES (continued)

LCO (continued)

unaffected steam generator. The block valves must be OPERABLE to isolate a failed open ADV line. A closed block valve does not render it or its ADV line inoperable if operator action time to open the block valve is supported in the accident analysis.

Failure to meet the LCO can result in the inability to cool the plant to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An ADV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand.

[For this facility, an OPERABLE ADV line constitutes the following:]

[For this facility, the support systems required OPERABLE to ensure the ADV lines OPERABILITY are as follows:]

[For this facility, those required support systems which, upon their failure, do not declare the ADV lines inoperable and their justification are as follows:]

APPLICABILITY

In MODE 1, 2, 3, or 4, the ADV lines provide the path for cooling the RCS to RHR entry conditions following an SGTR.

In MODE 5 or 6, an SGTR is not a credible event.

For this LCO, a Note has been added to provide clarification that all ADV lines are treated as an entity with a single Completion Time.

ACTIONS

A.1

With one ADV line inoperable, action should be taken to return the inoperable ADV line to OPERABLE status. The 7-day Completion Time takes into account the redundant capability afforded by the remaining OPERABLE ADV lines, a nonsafety grade backup in the Steam Bypass System, and

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BASES (continued)

ACTIONS (continued)

MSSVs. A Note has been added to this Required Action to indicate that the provisions of LCO 3.0.4 do not apply.

B.1

With more than one ADV line inoperable, action must be taken to restore at least [three] of the ADV lines to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the plant at power. The 24-hour Completion Time takes into account sufficient time to repair inoperable ADV lines, the availability of the Steam Bypass System, MSSVs, and the low probability of an event occurring during this period that requires the ADV lines.

C.1 and C.2

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 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 MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

SR 3.7.11.1 verifies the OPERABILITY of the ADVs. To perform a controlled cooldown of the RCS, the ADVs must be able to be opened and throttled through their full range. This SR ensures the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a plant cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.11.2

SR 3.7.11.2 verifies the OPERABILITY of the block valves. The function of the block valve is to isolate a failed open ADV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during plant cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. [Unit Name] FSAR, Section [10.3], "[Main Steam Supply System]."

B 3.7 PLANT SYSTEMS

B 3.7.12 Control Room Emergency Filtration System (CREFS)

BASES

BACKGROUND

The CREFS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The CREFS consists of two independent, redundant trains that recirculate and filter the control room air. Each train consists of a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Duct work, valves or dampers, and instrumentation also form part of the system, as well as demisters to remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines, and provide backup in case of failure of the main HEPA filter bank.

The CREFS is an emergency system, parts of which may also operate during normal plant operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the control room is isolated, and the stream of ventilation air is recirculated through the system's filter trains. The prefilters or demisters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the charcoal adsorbers.

Actuation of the CREFS places the system in either of two separate states (emergency radiation state, toxic gas isolation state) of the emergency mode of operation, depending on the initiation signal. Actuation of the system to the emergency radiation state of the emergency mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the control room air through the redundant trains of HEPA and the charcoal filters. The emergency

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BASES (continued)

BACKGROUND (continued)

radiation state also initiates pressurization and filtered ventilation of the air supply to the control room.

Outside air is filtered, [diluted with building air from the electrical equipment and cable spreading rooms,] and added to the air being recirculated from the control room. Pressurization of the control room prevents infiltration of unfiltered air from the surrounding areas of the building. The actions taken in the toxic-gas isolation state are the same, except that the signal switches control room ventilation to an isolation alignment to prevent outside air from entering the control room.

The air entering the control room is continuously monitored by radiation and toxic-gas detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic-gas isolation state, as required. The actions of the toxic-gas isolation state are more restrictive, and will override the actions of the emergency radiation state.

A single train will pressurize the control room to about [0.125] inches water gauge, and provide an air exchange rate in excess of [25%] per hour. The CREFS operation in maintaining the control room habitable is discussed in Reference 1.

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally-open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREFS is designed in accordance with Seismic Category I requirements.

The CREFS is designed to maintain the control room environment for 30 days continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5-rem whole body dose.

APPLICABLE SAFETY ANALYSES

The CREFS components are arranged in redundant, safety-related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

supply of filtered air to all areas requiring access. During emergency operation, the CREFS maintains the temperature between [70]°F and [85]°F. The CREFS 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 (LOCA), fission-product release presented in Reference 2.

The analysis of toxic-gas releases demonstrates that the toxicity limits are not exceeded in the control room following a toxic chemical release, as presented in Reference 1.

The worst-case single active failure of a component of the CREFS, assuming a loss-of-offsite power, does not impair the ability of the system to perform its design function.

The CREFS satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

Two independent and redundant trains of the CREFS are required to ensure that at least one is available assuming a single failure disables the other train. Total system failure could result in exceeding a dose of 5-rem to the control room operator in the event of a large radioactive release.

The CREFS is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both trains. A train is OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;

c. Heater, demister, duct work, valves, and dampers are OPERABLE, and air circulation can be maintained; and

d. SRs are met.

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BASES (continued)

LCO (continued)

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, duct work, and access doors.

[For this facility, the following support systems are required OPERABLE to ensure the CREFS OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not declare the CREFS inoperable and their justification are as follows:]

APPLICABILITY

In MODE 1, 2, 3, or 4, the CREFS must be OPERABLE to control operator exposure during and following a DBA.

[In MODE 5 or 6, the CREFS may be required to cope with the release from the rupture of an outside waste-gas tank.]

During movement of irradiated fuel, the CREFS must be OPERABLE to cope with the release from a fuel-handling accident.

ACTIONS

A.1

When one CREFS train is inoperable, it must be restored to OPERABLE status within 7 days. During this period, the remaining OPERABLE CREFS train is adequate to perform the control room protection function. The 7-day Completion Time is based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.

The concurrent failure of two CREFS 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, when Required Action A.1 cannot be completed within the required Completion Time, the plant must be placed in a MODE that minimizes accident risk. This is done by placing the plant in at least MODE 3 within

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BASES (continued)

ACTIONS (continued)

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

C.1

In MODE 5 or 6, or during movement of irradiated fuel when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CREFS train should be immediately placed in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure would be readily detected.

Required Action C.1 is modified by a Note to place the system in the emergency mode if auto-swapover to emergency mode is inoperable.

C.2.1, C.2.2, and C.2.3

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might enter the control room. This places the plant in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

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

When in MODE 5 or 6, and during movement of irradiated fuel with two CREFS trains inoperable, the Required Action is to immediately suspend activities that present a potential for releasing radioactivity that might enter the control room. This places the plant in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

SR 3.7.12.1 verifies that a train in a standby mode of operation starts on demand and continues to operate. Standby systems should be checked periodically to ensure

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

that they start and function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Systems without heaters need only to be operated for 15 minutes to demonstrate the function of the system. Furthermore, the 31-day Frequency was developed considering the known reliability of the equipment and the two-train redundancy availability.

SR 3.7.12.2

The Ventilation Filter Testing Program (VFTP) encompasses all the CREFS filter tests in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

SR 3.7.12.3 demonstrates that on an actual or simulated actuation signal(s) [including toxic-gas detector signal,] each CREFS train starts and operates. The Frequency of 18 months is specified in Regulatory Guide 1.52 (Ref. 3).

SR 3.7.12.4

SR 3.7.12.4 demonstrates the integrity of the control room enclosure, and the assumed inleaking rates of the potentially contaminated air. The control rooms positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CREFS. During the emergency mode of operation, the CREFS is designed to pressurize the control room to [0.125] inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The CREFS is designed to maintain this positive pressure with one train at a recirculation flow rate of [35,700] cfm. The Frequency of 18 months is consistent with the guidance provided in Reference 4.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	SR 3.7.12.5 [For this facility, the purpose of this SR is as follows:]
REFERENCES	1. [Unit Name] FSAR, Section [6.4], "[Habitability Systems]." 2. [Unit Name] FSAR, Section [15], "[Accident Analysis]." 3. Regulatory Guide 1.52 (Rev. 02), "Design, Testing and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants." 4. NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability System," Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.13 Control Room Emergency Air Temperature Control System (CREHVAC)

BASES

BACKGROUND

The CREHVAC provides temperature control for the control room following isolation of the control room.

The CREHVAC consists of two independent, redundant trains that provide cooling and heating of recirculated control room air. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for control room temperature control. The CREHVAC is a subsystem providing air temperature control for the Control Room Emergency Filtration System (LCO 3.7.12).

The CREHVAC is an emergency system, parts of which may also operate during normal plant operations. A single train will provide the required temperature control to maintain the control room between [70]°F and [85]°F. The CREHVAC operation in maintaining the control room temperature is discussed in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the CREHVAC is to maintain the control room temperature within limits for 30 days continuous occupancy.

The CREHVAC components are arranged in redundant, safety-related trains. During emergency operation, the CREHVAC maintains the temperature between [70]°F and [85]°F. A single-active failure of a component of the CREHVAC, with a loss-of-offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The CREHVAC is designed in accordance with Seismic Category I requirements. The CREHVAC is capable of removing sensible- and latent-heat loads from the control room, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

The CREHVAC satisfies Criterion 3 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

LCO

Two independent and redundant trains of the CREHVAC are required to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the control room becoming uninhabitable, and the equipment operating temperature exceeding limits in the event of an accident.

The CREHVAC is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the heating and cooling coils and associated temperature control instrumentation. In addition, the CREHVAC must be operable to the extent that air circulation can be maintained.

[For this facility, the following support systems are required to be OPERABLE to ensure the CREHVAC OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring the CREHVAC inoperable and their justification are as follows:]

APPLICABILITY

In MODE 1, 2, 3, or 4, and during movement of irradiated fuel, the CREHVAC must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

In MODE 5 or 6, CREHVAC may not be required for those facilities that do not require automatic control room isolation.

ACTIONS

A.1

With one train of CREHVAC inoperable, the inoperable CREHVAC train must be restored to OPERABLE status within 30 days. During this period, the remaining OPERABLE CREHVAC train is adequate to maintain the control room temperature within limits. The 30-day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the

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(continued)

BASES (continued)

ACTIONS (continued)

required protection, and that alternate safety or nonsafety related cooling means are available. [For this facility, the alternate cooling means are as follows:]

The concurrent failure of two CREHVAC 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, when Required Action A.1 cannot be completed within the required Completion Time, the plant must be placed in a MODE that minimizes the risk. This is done by placing the plant 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 MODES from full power in an orderly manner and without challenging plant systems.

C.1

In MODE 5 or 6, or during movement of irradiated fuel, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CREHVAC train should be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures that would prevent automatic actuation will occur, and that active failures would be readily detected.

C.2.1, C.2.2, and C.2.3

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the plant in a condition that minimizes risk, but does not preclude the movement of fuel to a safe position.

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

When in MODE 5 or 6, or during movement of irradiated fuel with two CREHVAC train inoperable, the Required Action is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the plant in a condition that

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BASES (continued)

ACTIONS (continued)	minimizes risk. This does not preclude the movement of fuel to a safe position.
SURVEILLANCE REQUIREMENTS	SR 3.7.13.1 SR 3.7.13.1 verifies that the heat removal capability of the system is sufficient to meet design requirements. The surveillance is performed at a Frequency of 18 months, because significant degradation of the CREHVAC is slow and is not expected over this time period. It consists of a combination of testing and calculations.
REFERENCES	1. [Unit Name] FSAR, Section [6.4], "[Habitability Systems]."

B 3.7 PLANT SYSTEMS

B 3.7.14 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

BASES

BACKGROUND

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a loss-of-coolant accident (LOCA). The ECCS PREACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area and the lower reaches of the auxiliary building.

The ECCS PREACS consists of two independent, redundant trains. Each train consists of a heater, a prefilter or demister, 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, as well as demisters functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the pump room following receipt of a safety injection (SI) signal.

The ECCS PREACS is a standby system, aligned to bypass the system HEPA filters and charcoal adsorbers. During emergency operations, the ECCS PREACS dampers are realigned, and fans are started to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the ECCS pump room isolate, and the stream of ventilation air discharges through the system's 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 ECCS PREACS is discussed in several sections of the FSAR (Refs. 1, 2, and 3) since it may be used for normal, as well

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BASES (continued)

BACKGROUND (continued)

as post-accident, atmospheric cleanup functions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level, consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 4).

APPLICABLE SAFETY ANALYSES

The design basis of the ECCS PREACS is established by the large-break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as a SI pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 CFR 100 limits (Ref. 5), or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The analysis of the effects and consequences of a large-break LOCA is presented in Reference 3. The ECCS PREACS also actuates following a small-break LOCA, requiring the plant to go into the recirculation mode of long-term cooling, and to clean up releases of smaller leaks, such as from valve stem packing.

Two types of system failures are considered in the accident analysis: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

The ECCS PREACS satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

Two independent and redundant trains of the ECCS PREACS are required 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 the atmospheric release from the ECCS pump room exceeding 10 CFR 100 limits in the event of a Design Basis Accident (DBA).

ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration are OPERABLE in both trains.

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BASES (continued)

LCO (continued)

A 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;

c. Heater, demister, ductwork, valves, and dampers are OPERABLE and air circulation can be maintained; and

d. SRs are met.

[For this facility, the following support systems are required to be OPERABLE to ensure ECCS PREACS OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not declare the ECCS PREACS inoperable and their justification are as follows:]

[For this facility, the main systems supported by ECCS PREACS and the justification for not declaring the main systems inoperable, upon failure of ECCS PREACS, are as follows:]

APPLICABILITY

In MODE 1, 2, 3, or 4, the ECCS PREACS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODE 5 or 6, the ECCS PREACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one ECCS PREACS train not OPERABLE, the inoperable train must be restored to OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS PREACS function.

The 7-day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72-hour

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BASES (continued)

ACTIONS (continued)

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 the consideration that the remaining train can provide the required capability.

Concurrent failure of two ECCS PREACS trains would result in the loss of functional capability; therefore, LCO 3.0.3 must be entered immediately.

B.1

With one ECCS PREACS train inoperable, verify that the Required Actions have been initiated for those supported systems declared inoperable by the support ECCS PREACS train within a Completion Time of [ ] hours.

The [ ]-hour Completion Time is defined as the most limiting of all the Required Actions for all the supported systems that needed to be declared inoperable upon the failure of one or more support features specified under Condition B.

Required Action B.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of ECCS PREACS trains have been initiated by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Conditions B of this LCO.]

[For this facility, the identified supported systems' Required Actions are as follows:]

C.1

With one ECCS PREACS train inoperable, and one or more required support or supported features inoperable associated with the other redundant ECCS PREACS train; a loss of function capability results, and LCO 3.0.3 must be entered immediately. However, if the support or supported features' LCOs take into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

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BASES (continued)

ACTIONS (continued)

D.1 and D.2

If the ECCS PREACS train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 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 MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

SR 3.7.14.1 verifies that a train in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and 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 without heaters need only be operated for 15 minutes to demonstrate the function of the system. Furthermore, the 31-day Frequency was developed considering the known reliability of equipment and the two-train redundancy available.

SR 3.7.14.2

Specification 5.7.4.p of the Ventilation Filter Testing Program (VFTP) encompasses all the ECCS PREACS filter tests in accordance with Regulatory Guide 1.52 (Ref. 4). 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.14.3

This SR 3.7.14.3 demonstrates that each ECCS PREACS train starts and operates on an actual or simulated actuation signal. The 18-month Frequency is consistent with that specified in Regulatory Guide 1.52 (Ref. 4).

SR 3.7.14.4

SR 3.7.14.4 demonstrates the integrity of the ECCS pump room enclosure. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the ECCS PREACS. During the emergency mode of operation, the ECCS PREACS is designed to maintain a slight negative pressure in the ECCS pump room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The ECCS PREACS is designed to maintain this negative pressure at a flow rate of [20,000] cfm from the ECCS pump room. The Frequency of 18 months is consistent with the guidance provided in Reference 6.

The minimum system flow rate maintains a slight negative pressure in the ECCS pump room area, and provides sufficient air velocity to transport particulate contaminants, assuming only one operating filter train.

The number of filter elements is selected to limit the flow rate through any individual element to about [1,000] cfm. This may vary, based on filter housing geometry. The maximum limit ensures that flow through, and pressure drop across, each filter element are not excessive.

The number and depth of the adsorber elements ensure that, at the maximum flow rate, the residence time of the air stream in the charcoal bed achieves the desired adsorption rate. At least a [0.125] second residence time is necessary for an assumed [99]% efficiency.

When clean, the filters have a ce[illegible] pressure drop at the design flow rate. The magnitude of the pressure drop indicates acceptable performance and is based on manufacturers' recommendations for the filter and adsorber

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

elements at the design flow rate. An increase in pressure drop, or a decrease in flow indicates that the filter is being loaded or that other problems exist within the system.

This test is conducted with the tests for filter penetration; thus, an 18-month Frequency, consistent with that specified in Regulatory Guide 1.52 (Ref. 4), is used.

SR 3.7.14.5

Operating the bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the bypass damper is verified if it can be opened. An 18-month Frequency is consistent with that specified in Regulatory Guide 1.52 (Ref. 4).

REFERENCES

1. [Unit Name] FSAR, Section [6.5.1], "[ESF Atmosphere Cleanup Systems]."

2. [Unit Name] FSAR, Section [9.4.5], "[Engineered Safety Feature Ventilation System]."

3. [Unit Name] FSAR, Section [15.6.5], "[Loss of Coolant Accidents]."

4. Regulatory Guide 1.52 (Rev. 2), "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

5. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

6. NUREG-0800, Section 6.5.1, "Standard Review Plan," Rev. 2, "ESF Atmosphere Cleanup System," July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Building Air Cleanup System (FBACS)

BASES

BACKGROUND

The FBACS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident or loss-of-coolant accident (LOCA). The FBACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

The FBACS consists of two independent, redundant trains. Each train consists of a heater, a prefilter or demister, 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, as well as demisters functioning to reduce the relative humidity of the airstream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal.

The FBACS is a standby system, parts of which may also be operated during normal plant operations. Upon receipt of the actuating signal, normal air discharges from the building, the fuel-handling building is isolated, and the stream of ventilation air discharges through the system's filter trains. The prefilters or demisters 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 FBACS is discussed in several sections of the FSAR (Refs. 1, 2, and 3) because it may be used for normal, as well as post-accident, atmospheric cleanup functions.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The FBACS 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 all fuel rods in an assembly are damaged. The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the FBACS. The DBA analysis of the fuel-handling accident assumes that only one train of the FBACS 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 one remaining train of this filtration system. The amount of fission products available for release from the fuel-handling building is determined for a fuel-handling accident and for a LOCA. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).

The FBACS satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

Two independent and redundant trains of the FBACS are required to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss-of-offsite power. Total system failure could result in the atmospheric release from the fuel-handling building exceeding the 10 CFR 100 limits in the event of a fuel-handling accident.

The FBACS is considered OPERABLE when the individual components necessary to control exposure in the fuel-handling building are OPERABLE in both trains. A 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 function;

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BASES (continued)

LCO (continued)

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained; and

d. SRs are met.

[For this facility, the following support systems are required OPERABLE to ensure the FBACS OPERABILITY are as follows:]

[For this facility, those required support systems which, upon their failure, do not declare the FBACS inoperable and their justification are as follows:]

APPLICABILITY

In MODE 1, 2, 3, or 4, the FBACS is required to be OPERABLE to provide fission-product removal associated with ECCS leaks due to a LOCA (refer to LCO 3.7.14), for plants that use this system as part of their ECCS Pump Room Exhaust Air Cleanup System (PREACS).

In MODE 5 or 6, the FBACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

During movement of irradiated fuel in the fuel building, the FBACS is always required to be OPERABLE to alleviate the consequences of a fuel-handling accident.

ACTIONS

A.1

If one FBACS train is inoperable, the inoperable train must be restored to OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the FBACS function. The 7-day Completion Time is based on the risk from an event requiring the inoperable FBACS train, considering that the remaining FBACS train can 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 required Completion Time, or when both

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ACTIONS (continued)

FBACS trains are inoperable, the plant must be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in MODE 3 within 6 hours, and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

C.1 and C.2

When Required Action A.1 cannot be completed within the required Completion Time during movement of irradiated fuel in the fuel building, the OPERABLE FBACS train should be started immediately or fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures that would prevent system operation will occur, and that any active failure 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 to a safe position.

D.1

When two trains of the FBACS are inoperable during movement of irradiated fuel in the fuel building, action should be taken to place the plant in a condition in which the LCO is not applicable. This LCO involves immediately suspending movement of irradiated fuel in the fuel building. This action does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

SR 3.7.15.1 demonstrates that a train in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Monthly heater operation drys out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems without heaters need only be operated for 15 minutes to demonstrate the function of the system.] Furthermore, the 31-day Frequency was developed considering the known reliability of the equipment and the two-train redundancy available.

SR 3.7.15.2

Specification 5.8.4.p of the Ventilation Filter Testing Program (VFTP) encompasses all the FBACS filter tests 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 3.7.15.3

SR 3.7.15.3 demonstrates that each FBACS train starts and operates on an actual or simulated actuation signal. The 18-month Frequency is consistent with that specified in Regulatory Guide 1.52 (Ref. 5).

SR 3.7.15.4

SR 3.7.15.4 demonstrates the integrity of the fuel building enclosure. The ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FBACS. During the emergency mode of operation, the FBACS is designed to maintain a slight negative pressure in the fuel building, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The FBACS is designed to maintain this negative pressure at a flow rate of [20,000] cfm to the fuel building. The Frequency of 18 months is consistent with the guidance provided in Reference 6.

The minimum system flow rate maintains a negative pressure in the fuel-handling building and provides sufficient air

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

velocity to transport particulate contaminants, assuming that only one filter train is in operation.

The number of filter elements is selected to limit the flow rate through any individual element to about [1,000] cfm. This may vary, based on filter housing geometry. The maximum limit ensures that flow through, and pressure drop across, each filter element are not excessive.

The number and depth of the adsorber elements ensure that, at the maximum flow rate, the residence time of the air stream in the charcoal bed achieves the desired adsorption rate. At least a [0.125]-second residence time is necessary for an assumed [99]% efficiency.

When clean, the filters have a certain pressure drop at the design flow rate. The pressure drop indicates acceptable performance and is based on manufacturers' recommendations for the filter and adsorber elements at the design flow rate. An increase in pressure drop, or a decrease in flow indicates that the filter is being loaded, or that other problems exist within the system.

This test is conducted with the tests for filter penetration; thus, an 18-month Frequency is consistent with that specified in Regulatory Guide 1.52 (Ref. 5), is used.

SR 3.7.15.5

Operating the bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the bypass damper is verified if it can be opened. An 18-month Frequency is consistent with that in Regulatory Guide 1.52 (Ref. 5).

REFERENCES

1. [Unit Name] FSAR, Section [6.5.1], "[ESF Atmosphere Cleanup Systems]."

2. [Unit Name] FSAR, Section [9.4.5], "[Engineered Safety Feature Ventilation System]."

3. [Unit Name] FSAR, Section [15.7.4], "[Fuel Handling Accident]."

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BASES (continued)

REFERENCES (continued)

4. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

5. Regulatory Guide 1.52 (Rev. 2), "Design, Testing and Maintenance Criteria for Post-Accident Engineered Safety-Feature Atmospheric Cleanup System Air Filtration and Absorption Units of Light-Water Cooled Nuclear Power Plants."

6. NUREG-0800, "Standard Review Plan," Section 6.5.1, Rev. 2, "ESF Atmosphere Cleanup System." July 1981

B 3.7 PLANT SYSTEMS

B 3.7.16 Penetration Room Exhaust Air Cleanup System (PREACS)

BASES

BACKGROUND

The PREACS filters air from the penetration area between containment and the auxiliary building.

The PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, 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, as well as demisters functioning to reduce the relative humidity of the airstream, also form part of the system. A second bank of HEPA filters, which follows the adsorber section, collects carbon fines and provides backup in case of failure of the main HEPA filter bank. The downstream HEPA filter, although not credited in the accident analysis, collects charcoal fines and serves as a backup should the upstream HEPA filter develop a leak. The system initiates filtered ventilation following receipt of a safety injection (SI) signal.

The PREACS is a standby system, parts of which may also operate during normal plant operations. During emergency operations, the PREACS dampers are realigned and fans are started to initiate filtration. Upon receipt of the actuating signal(s), normal air discharges from the penetration room, the penetration room is isolated, and the stream of ventilation air discharges through the system's filter trains. The prefilters remove any large particles in the air, as well as any entrained water droplets, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The PREACS is discussed in several sections of the FSAR (Refs. 1, 2, and 3) since it may be used for normal, as well as post-accident, atmospheric cleanup functions. Heaters may be included for moisture removal on systems operating in high humidity conditions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 4).

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The design basis of the PREACS is established by the large-break loss-of-coolant accident (LOCA). The system evaluation assumes a passive failure outside containment, such as valve packing leakage during a Design Basis Accident (DBA). In such a case, the system restricts the radioactive release to within the 10 CFR 100 limits, or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The analysis of the effects and consequences of a large-break LOCA are presented in Reference 3.

Two types of system failures are considered in the accident analysis: a complete loss of function, and excessive LEAKAGE. Either type of failure may result in less efficient removal of any gaseous or particulate material released to the penetration room following a LOCA.

The PREACS satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

Two independent and redundant trains of the PREACS are required to be OPERABLE to ensure that at least one train is available, assuming there is a single failure disabling the other train coincident with a loss-of-offsite power.

The PREACS is considered OPERABLE when the individual components necessary to control radioactive releases are OPERABLE in both trains. A 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;

c. Heater, demister, ductwork, valves, and dampers are OPERABLE and air circulation can be maintained; and

d. SRs are met.

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BASES (continued)

LCO (continued)

[For this facility, the following support systems are required to be OPERABLE to ensure the PREACS OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not declare the PREACS inoperable and their justification are as follows:]

[For this facility, the main systems supported by PREACS and the justification for not declaring the main systems inoperable upon failure of PREACS are as follows:]

APPLICABILITY

In MODE 1, 2, 3, or 4, the PREACS is required to be OPERABLE, consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).

In MODE 5 or 6, the PREACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one PREACS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the PREACS function. The 7-day Completion Time is appropriate because the risk contribution of the PREACS 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 period, and the fact that the remaining train can provide the required capability.

B.1

With one PREACS train inoperable, verify that the Required Actions have been initiated for those supported systems declared inoperable by the support PREACS train within a Completion Time of [ ] hours.

The [ ]-hour Completion Time is defined as the most limiting of all the Required Actions for all the supported systems

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BASES (continued)

ACTIONS (continued)

that needed to be declared inoperable upon the failure of one or more support features specified under Condition B.

Required Action B.1 ensures that those identified Required Actions associated with supported systems impacted by the inoperability of PREACS trains have been initiated by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition B of this LCO.]

[For this facility, the identified supported systems' Required Actions are as follows:]

C.1

With one PREACS train inoperable, and one or more required support or supported features inoperable associated with the other redundant PREACS train; a loss-of-function capability results, and LCO 3.0.3 must be entered immediately. However, if the support or supported features' LCOs take into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

D.1 and D.2

If the inoperable train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 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 MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

SR 3.7.16.1 demonstrates that a train in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions on this system are not severe, testing

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

each train once every month provides an adequate check on this system. Monthly heater operation drys out any moisture that may have accumulated in the charcoal as a result of humidity in the ambient air. [Systems without heaters need only be operated for 15 minutes to demonstrate the function of the system.] Furthermore, the 31-day Frequency was developed considering the known reliability of equipment, and the two train redundancy available.

SR 3.7.16.2

The Ventilation Filter Testing Program (VFTP) encompasses all the PREACS filter tests in accordance with Regulatory Guide 1.52 (Ref. 4). 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 3.7.16.3

This SR 3.7.16.3 demonstrates that each PREACS starts and operates on an actual or simulated actuation signal. The 18-month Frequency is consistent with that specified in Regulatory Guide 1.52 (Ref. 4).

SR 3.7.16.4

This SR 3.7.16.4 demonstrates the integrity of the penetration room enclosure. The ability of the penetration room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of PREACS. During the emergency mode of operation, the PREACS is designed to maintain a slight negative pressure at a flow rate of [ ] cfm in the penetration room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800 (Ref. 5).

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The minimum system flow rate maintains a slight negative pressure in the penetration room area, and provides sufficient air velocity to transport particulate contaminants, assuming only one filter train operating. The number of filter elements is selected to limit the flow rate through any individual element to about [1000] cfm. This may vary based on filter housing geometry. The maximum limit ensures that the flow through, and pressure drop across, each filter element are not excessive.

The number and depth of the adsorber elements ensure that, at the maximum flow rate, the residence time of the air stream in the charcoal bed achieves the desired adsorption rate. At least a [0.125]-second residence time is necessary for an assumed [99]% efficiency.

The filters have a certain pressure drop at the design flow rate when clean. The magnitude of the pressure drop indicates acceptable performance, and is based on manufacturers' recommendations for the filter and adsorber elements at the design flow rate. An increase in pressure drop, or a decrease in flow indicates that the filter is being loaded or that there are other problems with the system.

This test is conducted along with the tests for filter penetration. Thus, the 18-month Frequency is consistent with that specified in Regulatory Guide 1.52 (Ref. 4).

SR 3.7.16.5

It is necessary to operate the bypass damper to ensure that the system functions properly. The OPERABILITY of the bypass damper is verified if it can be opened. An 18-month Frequency is consistent with that specified in Regulatory Guide 1.52 (Ref. 4).

REFERENCES

1. [Unit Name] FSAR, Section [6.5.1], "[ESF Atmosphere Cleanup Systems]."

2. [Unit Name] FSAR, Section [9.4.5], "[Engineered Safety Feature Ventilation System]."

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BASES (continued)

REFERENCES (continued)

3. [Unit Name] FSAR, Section [15.6.5], "[Loss of Coolant Accidents]."

4. Regulatory Guide 1.52 (Rev. 02), "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

5. NUREG-800, Section 6.5.1, "Standard Review Plan," Rev. 2, "ESF Atmosphere Cleanup Systems," July 1981.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources—Operating

BASES

BACKGROUND

Introduction

The [Division 1] {VS-BW,CE,W,BWR/4: and [Division 2]} {VS-BWR/6: , [Division 2], and [Division 3]} AC source consist of the offsite power sources [preferred power sources, normal and alternate(s)], and the onsite standby power sources [[Division 1] {VS-BW,CE,W,BWR/4: and [Division 2]} {VS-BWR/6: , [Division 2], and [Division 3]} diesel generators]. As required by 10 CFR 50, Appendix A, GDC 17, "Electric Power Systems" (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.

{VS-BW,CE,W,BWR/4: The onsite Class 1E AC Distribution System supplies electrical power to [two redundant divisional load groups], with each [division] powered by [an independent Class 1E 4.16 kV ESF bus]. [Each [ESF bus] has at least [one] separate and independent offsite source[s] of power as well as a dedicated onsite diesel generator source.] The [Division 1 and Division 2] ESF systems each provide for the minimum safety functions necessary to shut down the unit and maintain it in a safe shutdown condition. [An electrical power distribution system diagram is provided in Figure B 3.8.1-1.]}

{VS-BWR/6: The onsite Class 1E AC Distribution System supplies electrical power to [three divisional load groups], with each [division] powered by an [independent Class 1E 4.16 kV ESF bus]. The [Division 1 and 2] [ESF buses] each have at least [one] separate and independent offsite source[s] of power. The [Division 3] [ESF bus] has at least [one] offsite source[s] of power. Each [ESF bus] has a dedicated onsite diesel generator. The ESF systems of any two of the three [divisions] provide for the minimum safety functions necessary to shut down the unit and maintain it in a safe shutdown condition. [An electrical power distribution system diagram is provided in Figure B.3.8.1-1.]}

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(continued)

BASES (continued)

"This Figure For Illustration Only. Do Not Use For Operation"

[PLANT SPECIFIC FIGURE]

[inclusion optional]

Figure B 3.8.1-1 (Page 1 of 1)
Electrical Power System

(continued)

(continued)

BASES (continued)

BACKGROUND (continued)

The redundant parts of the AC electrical power system are electrically, physically, and functionally independent to the extent that no single failure will cause a total loss of power to redundant safety-related load groups.

A single failure is an occurrence that results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Electrical systems are considered to be designed against an assumed single failure if neither a single failure of any active component (assuming passive components function properly) nor a single failure of a passive component (assuming active components function properly) results in a loss of the capability of the system to perform its safety functions.

In the event of a loss of preferred power, the ESF switchgears are automatically connected to the diesel generators in sufficient time for safe reactor shutdown or in sufficient time to mitigate the consequences of a Design Basis Accident (DBA) such as a loss-of-coolant accident (LOCA).

Offsite Sources

Offsite power is supplied to the [plant name] [switchyard(s)] from the transmission network by [two] transmission lines, which come into [the switchyard(s) via [two] right-of-way(s)]. 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]. The [two] offsite AC electrical power sources are designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions.

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 or buses].

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BASES (continued)

BACKGROUND (continued)

[PLANT SPECIFIC:
Provide description of any other salient features of the offsite power sources. Items that may be covered include:

a. Circuit breakers and protective relaying;

b. Ability to cross tie offsite circuits so that one circuit may power both ESF buses;

c. Normal at-power and shutdown electrical alignments;

d. Offsite circuit capability;

e. Ability to power ESF buses from the plant's own generator output via the unit auxiliary transformers; and

f. A description, for both the at-power and shutdown lineups, of alternate power availability from alternate offsite power circuits. Include in the description the capability of the alternate circuits, and whether the circuit is immediate or delayed access. If it is a delayed access circuit, describe what has to be done to gain access to the circuit (such as remove generator disconnect links) and whether the actions can be done remotely from the control room. Also state the amount of time required to perform the actions.

g. Discuss whether the sequencer is a support system for the offsite circuits, and whether the circuits are block-loaded with ESF loads, or whether they have the loads sequenced onto them.

h. Define and discuss the physical and functional characteristics of the offsite circuits that make them "separate and independent." Also, "separate" should be defined in terms of firedoors not closed, etc.]

Onsite Sources

The onsite standby power source for each [4.16 kV ESF bus] is a dedicated diesel generator. {VS-BW,CE,W,BWR/4: [Diesel generators (DGs) [11] and [12] are dedicated to ESF buses

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BASES (continued)

BACKGROUND (continued)

[11] and [12], respectively].} {VS-BWR/6: [Diesel (DGs) generators [11], [12], and [13] are dedicated to ESF buses [11], [12], and [13], respectively].} A DG starts automatically on {VS-BW,CE,W: [a safety injection signal (SIS) (i.e., low pressurizer pressure or high containment pressure signals)]} {VS-GE:[a LOCA signal (i.e., low reactor water level signal or high drywell pressure signal)]} or on an [ESF bus degraded voltage or undervoltage signal]. The undervoltage trip device senses a severe loss-of-voltage to a level at which electrical equipment would not function. The degraded voltage trip device senses a loss of voltage condition at which the equipment would function, but would sustain damage and become inoperable if operated for extended periods with degraded voltage. Additionally, after the diesel generator 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 signal. The DGs will also start and operate in the standby mode without tieing to the [ESF bus] on a safety injection signal alone. Following the trip of offsite power, a sequencer strips all non-permanent loads from the [ESF bus]. When the DG is tied to the [ESF bus], loads are then sequentially connected to their respective [ESF bus] by their automatic sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent an overburdened DG by automatic load application.

Ratings for [Division 1] {VS-BW,CE,W,BWR/4: and [Division 2]} {VS-BWR/6: , [Division 2], and [Division 3]} DGs satisfy the requirements of Regulatory Guide 1.9, "Selection, Design, and Qualification of DG Units Used as Onsite Electric Power Systems at Nuclear Power Plants" (Ref. 2). The continuous service rating of each of the DGs is [7,000] kW for [Divisions 1 and 2] {VS-BWR/6: and is [3,000] kW for [Division 3]} 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 3.

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BASES (continued)

BACKGROUND (continued)

Automatic Sequencers

The sequencer(s) is (are) activated by one of two conditions, [ESF bus] undervoltage (UV) or {VS-BW,CE,W: SIS} {VS-GE: LOCA signal}. Upon receipt of either or both of the initiating signals, the following actions will take place:

a. The DGs start;

b. Any test sequence in progress stops;

c. The [ESF bus] of all non-permanent loads (UV only) is stripped;

d. The DG breaker (UV only) closes; and

e. The appropriate loads as determined by the initiating signal energize.

Required plant loads are returned to service in a sequence determined to ensure that the most essential loads are started first while preventing overloading of the DGs in the process. Within [1 minute] after the initiating signal is received, all loads needed to recover the plant or maintain it in a safe condition are returned to service.

The sequencer 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's sequencer] affects every major ESF system in the [division].

APPLICABLE SAFETY ANALYSES

The initial conditions of design basis transient and accident analyses in the FSAR, [Chapter 6, "Engineering Safety Features"], and [Chapter 15, "Accident Analyses"], assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS) and containment design limits are not exceeded. These limits are discussed in more

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	detail in the Bases for Technical Specifications (TS) 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 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 plant. This includes maintaining at least one [division] of the onsite or offsite AC sources, DC power sources and associated distribution systems 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.

AC sources satisfy the requirements of Criterion 3 of NRC Interim Policy Statement.

LCO

As a minimum, the following AC electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System {VS-BWR/6: and a third [Division 3] circuit, not necessarily separate and independent from the first two}; and

b. {VS-BW,CE,W,BWR/4: Two} {VS-BWR/6: Three} separate and independent DGs
{VS-BW,CE,W,BWR/4: [11] and [12]}
{VS-BWR/6: [11], [12, and [13]}, each with:

1. separate day [and engine-mounted] fuel tanks containing a minimum volume of fuel within the limits specified in SR 3.8.1.8,

2. a separate Fuel Storage System containing a minimum volume of fuel within the limits specified in SR 3.8.1.9,

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BASES (continued)

LCO (continued)

3. a separate fuel transfer pump capable of meeting SR 3.8.1.16,

4. lubricating oil storage containing a minimum total volume of lubricating oil within the limits specified in SR 3.8.1.10,

5. capability to transfer lubricating oil from storage to the DG unit, and

6. separate air-start receivers containing a minimum air pressure within the limits of SR 3.8.1.7.

In addition, [one required automatic load sequencer per ESF bus] shall be OPERABLE. {VS-BWR/6: [PLANT SPECIFIC: In general, [Division 3] does not have a load sequencer since it has only one large load, i.e., high pressure core spray (HPCS) pump. In such cases the LCO should refer to the [Division 1 and 2] sequencers only.]}

For the offsite circuits, DGs, and sequencers to be OPERABLE, they must be capable of performing their intended function, have all support systems OPERABLE, and have successfully completed all SRs.

[Each facility will define what constitutes an OPERABLE offsite circuit, including the components of the circuit, such as breakers, transformers, switches, interrupting devices, protective relays, cabling and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF buses.]

[For this facility, as a minimum, the following support systems are required OPERABLE to assure offsite circuit OPERABILITY:]

[ ]

Inoperability of any of the offsite circuit support systems results immediately in an inoperable offsite circuit as per the definition of OPERABILITY; however, exceptions are

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BASES (continued)

LCO (continued)

allowed for specific support systems, provided that a justification is given. Therefore, upon the inoperability of the following support systems for an offsite circuit, the declaration of an inoperable offsite circuit may be delayed:

[ ]

The justification for delaying the declaration of offsite circuit inoperability for each of the above items is as follows:

[ ]

[Each facility will define what constitutes an OPERABLE DG, including the components of the DG, such as the diesel engine, generator, Fuel Storage System, starting and control air, combustion air intake and exhaust, cooling system, lubricating oil, ventilation, and DG output breaker.]

[For this facility, as a minimum, the following support systems are required OPERABLE to assure DG OPERABILITY:]

[ ]

Inoperability of any of the DG support systems results immediately in an inoperable DG as per the definition of OPERABILITY; however, exceptions are allowed for specific support systems provided that a justification is given. Therefore, upon the inoperability of the following support systems for a DG, the declaration of an inoperable DG may be delayed:

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BASES (continued)

LCO (continued)

[ ]

The justification for delaying the declaration of DG inoperability for each of the above items is as follows:

[ ]

[Each facility will define what constitutes an OPERABLE [automatic sequencer, including the components of the sequencer such as programmable logic arrays].

[For this facility, as a minimum, the following support systems are required OPERABLE to assure [automatic sequencer] OPERABILITY:]

[ ]

Inoperability of any of the [automatic sequencer] support systems results immediately in an inoperable [automatic sequencer] as per the definition of OPERABILITY; however, exceptions are allowed for specific support systems provided that a justification is given.

Therefore, upon the inoperability of the following support systems for an [automatic sequencer], the declaration of an inoperable [automatic sequencer] may be delayed:

[ ]

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(continued)

BASES (continued)

LCO (continued)

The justification for delaying the declaration of [automatic sequencer] inoperability for each of the above items is as follows:

[ ]

AC Sources and Component OPERABILITY

The definition of OPERABILITY states that a component shall be OPERABLE when it is capable of performing its specified functions and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the component to perform its functions are also capable of performing their related support functions. When applying this definition to a component, say an Emergency Core Cooling System (ECCS) pump, the question arises, "How many AC sources are necessary for the pump to be considered OPERABLE?" For the electrical power distribution buses to be OPERABLE, they simply have to be fully energized by one of the capable sources accepted in the plant design, within design voltage and frequency tolerances, and within allowable environmental parameters such as temperature and humidity. Similarly, an ECCS pump is OPERABLE if it is powered from such a fully energized and OPERABLE distribution system. Note that for OPERABILITY of both the distribution system and the components, no requirements, beyond at least one of the electrical power sources that was accepted as a part of the plant design, are made on how many electrical power sources are available to power the bus.

Thus, for plant components and distribution buses, zero electrical power sources means the component or bus is inoperable. Fully energized from at least one power source that was accepted as a part of the plant design means the component or bus is OPERABLE (at least from the point of view of needing electrical support). Thus, the principle for component (including electrical bus) OPERABILITY is that a component may be considered OPERABLE if it has electricity at its terminals (and the electricity came from a source that was accepted as a part of the plant design).

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BASES (continued)

LCO (continued)

With this interpretation of component OPERABILITY, the next question that arises is, "How can an ECCS pump that is only powered from an offsite source be considered OPERABLE?" If such a pump does not have electrical support from a DG, it will not be able to function given a DBA and a loss of offsite power. The short answer to this question is that it is not the ECCS pump that was broken in the above scenario. It was a DG that was inoperable. Thus, for operating MODES, this LCO 3.8.1 contains the necessary ACTIONS for an inoperable required AC source (including a DG). Similarly, for shutdown modes, LCO 3.8.2 contains the necessary ACTIONS for an inoperable required AC source under shutdown conditions. Cascading the inoperability of a single AC source (including DG) to every component in the [division] served by the AC source is not necessary. The longer answer to this question requires some additional explanation.

The electrical power systems at nuclear power plants are designed to meet the GDC listed in Appendix A of 10 CFR 50. The AC electrical power system is 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 containment design limits are not exceeded. The OPERABILITY of the power sources are based upon meeting the design basis of the plant. This includes maintaining at least:

a. {VS-BW,CE,W,BWR/4: One [division] ([Division 1 or Division 2])} {VS-BWR/6: Two out of three [divisions]} of the offsite AC and onsite DC power sources and associated distribution systems OPERABLE during accident conditions, assuming a loss of all onsite power and a single failure; and

b. {VS-BW,CE,W,BWR/4: One [division] ([Division 1 or Division 2])} {VS-BWR/6: Two out of three [divisions]} of the onsite AC and DC power sources and associated distribution systems OPERABLE during accident conditions, assuming a loss of all offsite power and a single failure.

See, for example, GDC 17, 33, 34, 35, 38, and 41.

An important corollary to or consequence of the design requirements (a) and (b) above is the following. For a

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BASES (continued)

LCO (continued)

safety-related component to be considered operable, it must have both a source of offsite and onsite power. This is the design basis definition that is shown here in lower case letters and underlined to distinguish it from the actual definition of OPERABLE that is used in the Technical Specifications. This definition of operable is every bit as valid as the design criteria for a nuclear plant. The difference is that a component is OPERABLE if it has at least one AC source; however, it may not be operable. To be operable, the component would have to have both an onsite and offsite AC source.

Let's examine the differences between OPERABLE and operable for the operating MODES of Applicability that are governed by Specification 3.8.1 (and other operating Technical Specifications). For a typical plant, the LCO of Specification 3.8.1 requires a DG and an offsite circuit for each [division]. Thus, as long as the LCO of Specification 3.8.1 is met, all components are both OPERABLE and operable (in terms of the electrical support they require). Furthermore, if three or more AC sources are inoperable, then the plant must enter LCO 3.0.3 and shut down. Therefore, in these two extremes, any difference between OPERABLE and operable becomes irrelevant. If two AC sources are inoperable on the same bus, and if that bus has no other source of power (e.g., a dead bus), then the two definitions also give the same result, and every component in the [division] is inoperable since they have no electrical power. In fact, the only time the difference becomes relevant is when one or two (but with no dead bus) AC sources become inoperable.

Thus, when in the ACTIONS of Specification 3.8.1 for one or two AC sources inoperable, the components in the [divisions] associated with the inoperable AC source(s) are generally OPERABLE but not operable. At this point, the reason for defining OPERABILITY as requiring only one AC source becomes clear. If one uses the design basis definition of operability in place of OPERABILITY, then every component in the [division] would have to be declared not operable upon the loss of a single AC source.

Performing the Required Actions of the TS for each component that requires AC power in a [division] (when the components still have AC power) just because one AC source is

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BASES (continued)

LCO (continued)

inoperable is not necessary. Fix the AC source and leave the components alone.

If we use the definition of <u>operability</u>, then upon the loss of two AC sources in different [divisions] the plant would have to enter LCO 3.0.3 since two entire safety [divisions] of components would be not <u>operable</u>. This would make the 2-, 12-, and 24-hour Completion Times specified in LCO 3.8.1 for two DGs inoperable, one DG and one offsite circuit inoperable, and two offsite circuits inoperable, respectively, irrelevant.

By not cascading the inoperability of a single AC source down to all the components in its safety [division], two things are lost:

a. The Required Actions for an inoperable component in the component LCO; and

b. A message to the component LCO that the component in this [division] is potentially inoperable under certain Design Basis Events.

The loss of (a) is probably not important. Usually, the Required Action is simply to restore the component to OPERABLE status. In this case, it is not the component that is broken, it is the AC source. The AC source will be fixed within its Completion Time, or other remedial actions, such as a plant shutdown, will be taken.

The loss of (b) *is* important. Most component LCOs do not allow continued plant operation with a complete loss of function. For example, a typical ECCS Specification will allow loss of ECCS function in one [division] for 72 hours but will require a shutdown if all ECCS function is lost. It is clear that if the design basis definition of <u>operability</u> was used, and if a DG in one [division] was out of service coincident with an ECCS pump in another [division], a shutdown would be required by the ECCS Specification since two ECCS pumps would be not <u>operable</u>. However, when the Specification definition of OPERABILITY is used in place of <u>operability</u>, the ECCS Specification shows one pump inoperable with a 72-hour Completion Time, and the AC sources TS would have one DG inoperable with a 72-hour

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BASES (continued)

LCO (continued)

Completion Time. Thus, there appears to be a difficulty if an AC source is out of service in one [division], and a required feature (such as an ECCS pump) is out of service in another [division].

The problem is that this situation (AC source inoperable in one [division], required feature inoperable in another) represents a potential loss of required feature function under some of the conditions set forth in the design basis. By using the TS definition of OPERABILITY, no message is sent to the required feature LCO upon the inoperability of an AC source. To fix this problem, a cross-[division] check is incorporated into this LCO 3.8.1. See Condition B (for offsite source inoperability) and see Condition D (for onsite source inoperability). The purpose of these two conditions is to recognize that when in them, the plant is in a potential loss-of-function situation. The effect of these two Conditions is to reduce the Completion Time for an inoperable AC source to less than 72 hours. See the appropriate ACTIONS discussion for more information.

Another point of view is that, in practice, the design basis requirement for <u>operability</u> is relaxed for brief periods of time (typically 72 hours or less) while in an AC Sources—Operating ACTION statement. If a [Division 1] DG is out of service, all of the components in the safety [division] associated with that DG are not declared inoperable (even though by the strict definition of <u>operability</u> above, they are, in fact, not <u>operable</u>). Instead, the definition of <u>operability</u> is relaxed to that of OPERABILITY, which says that if a component in the [division] that has an out-of-service DG has electricity at its terminals, it is OPERABLE for the purpose of satisfying its component LCO. Thus, the only ACTION that has to be taken is that of the DG LCO. This relaxation of the design basis definition of <u>operability</u> is deemed acceptable because the DG inoperability is only allowed to persist for a limited amount of time (e.g., 72 hours in this case). The net effect of this interpretation is that during the 72 hours, the GDC are not met. The plant could not take a worst-case single failure and still maintain all safety functions with a loss of all offsite AC sources. In other words, we accept the risk of loss of single-failure protection for an event that involves total loss of offsite AC sources for 72 hours.

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BASES (continued)

LCO (continued)

The above discussion holds equally well for the companion Condition of one offsite circuit inoperable (instead of a DG). Thus, the requirement for both an onsite and offsite AC source of power found in the definition of operability is relaxed for 72 hours while in the AC Sources—Operating ACTION statement for one offsite circuit inoperable.

This relaxation of a design basis requirement is only implemented when in an ACTION of Specification 3.8.1. At all other times, the correct design basis interpretation of the "Necessary electrical power" in the definition of operability is that both onsite and offsite AC sources are required for a component to be considered operable and thus meet the design basis requirements.

Separation and Independence of AC Sources

An additional corollary to or consequence of the design requirements in GDC 17 is that the AC sources in one [division] must be separate and independent (to the extent possible) of the AC sources in the other [division(s)]. For the onsite diesel generators, the separation and independence is complete. That is, GDC 17 requires,

> "The onsite electric power supplies, including the batteries, and the onsite electrical distribution system, shall have sufficient independence, for redundancy, and testability to perform their safety functions assuming a single failure."

For the offsite AC sources, the separation and independence is to the extent practical. That is, GDC 17 requires,

> "Electric power from the transmission network to the onsite electrical distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions."

It is not acceptable to extrapolate from these words in GDC 17 that the offsite circuits are not completely separate and independent and conclude therefore that a single circuit

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BASES (continued)

LCO (continued)	cross-tied between [divisions] meets the GDC 17 requirements for offsite sources. Similarly, if interrupting devices or protective relaying that normally serves to provide electrical independence between the two circuits are inoperable, it is not acceptable to conclude that all offsite circuits are still OPERABLE. In general, the two offsite circuits are to be maintained separate and independent to the same extent as in the plant design.

APPLICABILITY

The AC sources and sequencers are required to be OPERABLE in {VS-BW,CE,W: MODES 1, 2, 3, and 4} {VS-GE: MODES 1, 2, and 3} to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of anticipated operational occurrences (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.

AC power requirements for {VS-BW,CE,W: MODES 5 and 6} {VS-GE: MODES 4 and 5} are covered in Specification 3.8.2, "AC Sources—Shutdown."

A Note has been added to provide clarification that for this LCO, all required [Division 1] {VS-BW,CE,W,BWR/4: and [Division 2]} {VS-BWR/6: , [Division 2], and [Division 3]} AC electrical sources and [automatic sequencers] shall be treated as an entity with a single Completion Time.

ACTIONS

A.1

Condition A is one required offsite circuit inoperable. The Required Action A.1 is to restore all required AC electrical power sources (offsite circuits and DGs) to OPERABLE status within a Completion Time of 72 hours {VS-BWR/6: for [Division 1 and Division 2] and within [72 hours] for [Division 3]}.

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BASES (continued)

ACTIONS (continued)

Performance of SR 3.8.1.1 ensures a highly reliable power source and no common cause failure for the remaining required offsite {VS-BW,CE,W,BWR/4: circuit} {VS-BWR/6: circuits}. The OPERABILITY of the remaining required offsite {VS-BW,CE,W,BWR/4: circuit} {VS-BWR/6: circuits} must be verified once within 1 hour and once per 8 hours thereafter until the inoperable offsite circuit is restored to OPERABLE status.

SR 3.8.1.1 is only required when in Condition A. SR 3.8.1.1 is essentially identical to the normal weekly SR of offsite circuits (i.e., SR 3.8.1.4). The only difference is that SR 3.8.1.1 has a shorter Frequency for verification of the OPERABILITY of the remaining required OPERABLE offsite circuit. If a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition F, for two offsite circuits inoperable, is entered.

Per Regulatory Guide 1.93, "Availability of Electric Power Sources" (Ref. 4), operation may continue in Condition A for a period that should not exceed 72 hours {VS-BWR/6: for [Divisions 1 and 2]. The [72-hour] Completion Time for a [Division 3] offsite circuit inoperability is plant specific. Items to be considered in choosing this Completion Time are:

a. Potential light-loading of the [Division 3] DG during the [72-hour] period when the one required offsite circuit for [Division 3] is inoperable; and

b. The safety function of [Division 3].

In particular, the Completion Time for a [Division 3] offsite circuit inoperability shall not exceed 72 hours if [Division 3] systems support other ESF functions in addition to the HPCS function}. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the plant safety systems. In this Condition, however, the remaining OPERABLE offsite {VS-BW,CE,W,BWR/4: circuit} {VS-BWR/6: circuits} and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

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BASES (continued)

ACTIONS (continued)

The 72-hour {VS-BWR/6: (or 72-hour] for [Division 3]}) limit takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period. If Required Action A.1 and its associated Completion Time are not met, a controlled shutdown must be performed per Required Action J.1 and Required Action J.2.

{VS-BW,CE,W: B.1, B.2.1, and B.2.2}
{VS-GE: B.1, and B.2}

{VS-BW,CE,W:
Condition B is no offsite power to one [division] of the onsite Class 1E Power Distribution System AND one or more required support or supported features, or both, inoperable that are associated with the other [division] that has offsite power, or with opposite OPERABLE DC power subsystem(s), or both, OR the turbine-driven auxiliary feedwater pump inoperable.

{VS-W,CE,W:
Note that the OR in Condition B is not an exclusive "or". That is, the OR in Condition B includes Conditions in which:

a. One or more required support or supported features, or both, are inoperable. . .; or

b. A Condition in which the turbine-driven auxiliary feedwater pump is inoperable; or

c. Both (a) and (b) above.}

{VS-BWR/4:
Condition B is no offsite power to one [division] of the onsite Class 1E Power Distribution System AND one or more required support or supported features, or both, inoperable that are associated with the other [division] that has offsite power, or with opposite OPERABLE DC power subsystem(s), or both.}

{VS-BWR/6:
Condition B is no offsite power to one [division] of the onsite Class 1E Power Distribution System AND one or more required support or supported features, or both, inoperable that are associated with the other [divisions] that have

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BASES (continued)

ACTIONS (continued)

offsite power, or associated with opposite OPERABLE DC power subsystem(s), or both.}

Condition B is a companion Condition to Condition A. That is, it is not possible to be in Condition B without also being in Condition A. [For there to be no offsite power to one [division] of the onsite Class 1E Distribution System, one offsite circuit and any cross-ties to other offsite circuits must be inoperable or not connected.]

The rationale behind Condition B comes from GDC 33, 34, 35, 38, and 41. They state that,

> "Suitable redundancy in components and features, and suitable interconnections, leakage detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure."

If, as per the GDC, we assume that all onsite power is not available, then Condition B represents a loss of function for the feature that is inoperable in the other {VS-BW,CE,W,BWR/4: [division] that has} {VS-BWR/6: [divisions] that have} offsite power, or is associated with opposite OPERABLE DC power subsystem(s), or both.

Definition of BX: The allowable time for continued plant operation in Condition B is BX hours. BX is determined as follows. Consult the TS for the required feature that is inoperable. Define BXi as the Completion Time that the inoperable required feature TS allows for a complete loss of all required feature function. If no loss of function is allowed (e.g., if upon the loss of required feature function a shutdown is required), then assign BXi = 0 hours. For each required feature that is inoperable, there will be a BXi. BX is then defined as the minimum of all the BXi; however, if BX is found to be less than 24 hours, BX is reset to 24 hours. If BX is found to be greater than 72 hours, then BX is 72 hours.

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BASES (continued)

ACTIONS (continued)

There is one exception to the above rule for finding BX. Usually, 24 hours ≤ BX ≤ 72 hours. However, if the plant is in Condition B and Condition F (two required offsite circuits inoperable) simultaneously, then BX = 12 hours. The rationale for the reduction to 12 hours is that Condition F (two required offsite circuits inoperable) is assigned a Completion Time of 24 hours consistent with Regulatory Guide 1.93 (Ref 4.). However, on a risk basis, Regulatory Guide 1.93 allowed a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety [divisions] of components are OPERABLE. When in Condition B and F simultaneously, this is not the case, and a shorter Completion Time of BX = 12 hours is appropriate.

BX as defined above is acceptable because it minimizes risk while allowing time for restoration before subjecting the plant to transients associated with shutdown. (The above addresses the potential for loss of function under certain Conditions postulated in the design basis. In the event of an actual loss of function, the TS covering that loss of function will control the Completion Time.)

The specific list of "required support and supported features" encompassed by Condition B is provided in Reference 5. Required features are those that are designed with functionally redundant safety-related [divisions]. If a plant has a required feature that has no functionally redundant counterpart, that feature may not be required to be included. This is unlikely, however, since single-failure considerations usually require functional redundancy of safety features. Since the Completion Time allowance for this Required Action is limited to 72 hours, those systems with allowed Completion Times ≥ 72 hours for complete loss of function are not included as required features to be checked.

The reason that Condition B is for no offsite power to one [division] of the onsite Class 1E Distribution System is because losing one offsite circuit may not necessarily result in the total loss of offsite power to the [division] because of possible cross-ties to other offsite circuits. No offsite power source to one [division] needs to be established before the determination can be made whether an

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BASES (continued)

ACTIONS (continued)

inoperable redundant feature in the other [divisions] would result in a potential loss of function.

{VS-BW,CE,W:
Auxiliary feedwater is provided by a [50%]-capacity motor-driven feedwater pump in [Division 1], a [50%]-capacity motor-driven feedwater pump in [Division 2], and a [100%]-capacity turbine-driven feedwater pump. Therefore, assuming that all onsite power is not available (as per the GDC), Condition B reduces the 72-hour Completion Time to BX hours (see above for definition of BX) for the case in which auxiliary feedwater function has been reduced to only [50%] of capacity or less.}

{VS-BW,CE,W:
The turbine-driven auxiliary feedwater pump is not included with the "one or more required support or supported features, or both, inoperable that are associated with the other [division] that has offsite power," because the feedwater pump is steam driven (as opposed to motor driven), and thus is not "associated" with either [division] of the AC electrical power sources.}

{VS-BW,CE,W:
The Note for Required Action B.2.2 states, "Required Action B.2.2 is only required in MODES 1, 2, and 3, and in MODE 4 when auxiliary feedwater is being used for plant shutdown and startup." This Note is consistent with the Applicability requirements of Specification 3.7.4, "Auxiliary Feedwater System." When the pressure is < 715 psig] the turbine-driven auxiliary feedwater pump need not be capable of meeting the SR limits of SR 3.7.4.2 on developed head to satisfy the OPERABILITY requirements of Required Action B.2.2. The pump must be capable of coming up to speed and delivering flow, however. Furthermore, the licensee shall verify that the pump passed its last SR 3.7.4.2.}

Operation may continue in Condition B for a period that should not exceed BX hours. In this condition, the remaining OPERABLE offsite circuit and DGs are adequate to

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BASES (continued)

ACTIONS (continued)

supply electrical power to [Division 1 and Division 2] of the onsite Class 1E Distribution System. The BX-hour limit takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Thus, on a component basis, we may have lost single-failure protection for the required feature's function; however, we have not lost function. Similarly, we take into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period. If the Required Actions of Condition B and the associated Completion Times are not met, a controlled shutdown must be performed per Required Action J.1 and Required Action J.2.

C.1

Condition C is one required DG inoperable. Required Action C.1 is to restore the required AC electrical power sources (offsite circuits and DGs) to OPERABLE status within a Completion Time of 72 hours
{VS-BWR/6: for [Division 1 and Division 2] and within [72 hours] for [Division 3]}.

Performance of SR 3.8.1.2 ensures a highly reliable power supply by checking on the OPERABILITY of the required offsite circuits. SR 3.8.1.2 must be performed once within 1 hour of entering Condition C and once per 8 hours thereafter. Failing to perform SR 3.8.1.2 on a given circuit results in an inoperable circuit. Similarly, if a circuit fails to pass SR 3.8.1.2, it is inoperable. Upon offsite circuit inoperability, additional Conditions must then be entered to reflect the new plant state.

Performance of SR 3.8.1.3 ensures no common cause failure for the remaining required DG[s]. The determination of no common cause inoperability of the remaining required DG[s] must be made once within [8] hours of entering Condition C. If during the performance of SR 3.8.1.3 common cause is found, or if a required DG fails SR 3.8.1.3 for some other reason, then two required DGs are inoperable and Condition G is entered.

Note 3 of Condition C requires that SR 3.8.1.3 shall be completed if Condition C is entered. The intent is that all DG inoperabilities must be investigated for common cause

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BASES (continued)

ACTIONS (continued)

failures as per SR 3.8.1.3, regardless of how long the DG inoperability persists.

Per Regulatory Guide 1.93, "Availability of Electric Power Sources" (Ref. 4), operation may continue in Condition C for a period that should not exceed 72 hours {VS-BWR/6: for [Divisions 1 and 2]. The [72-hour] Completion Time for a [Division 3] DG inoperability is based upon the risk-significance of the [Division 3] DG in coping with a station blackout (SBO). Calculations show that the core melt frequency increases substantially for an SBO with a [Division 3] DG inoperable for 14 days as compared to an SBO with an OPERABLE [Division 3] DG.

The Completion Time for a [Division 3] DG may be increased from [72 hours] to [14 days] consistent with the HPCS TS provided:

a. The [Division 3] sole function is to support the HPCS function; and

b. Calculations show that the increase in the core melt frequency for an SBO with an inoperable [Division 3] DG is acceptably low.

If other ESF functions are supported by [Division 3], or if calculations show that the increase in core melt frequency for an SBO with an inoperable [Division 3] DG is unacceptably high, then the Completion Time for an inoperable [Division 3] DG shall be [72 hours].}

In Condition C, the remaining OPERABLE DG[s] and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 72-hour {VS-BWR/6: (or [72-hour] for [Division 3])} limit takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period. If Required Action C.1 and its associated Completion Time are not met, a controlled shutdown must be performed per Required Action J.1 and Required Action J.2.

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BASES (continued)

ACTIONS (continued)

{VS-BW,CE,W: D.1, D.2.1, and D.2.2}
{VS-GE: D.1 and D.2}

{VS-BW,CE,W:
Condition D is one required DG inoperable AND one or more required support or supported features, or both, inoperable that are associated with the OPERABLE DG[s], or with an opposite OPERABLE DC power subsystem, or both, OR the turbine-driven auxiliary feedwater pump inoperable.

{VS-BW,CE,W:
Note that the OR in Condition D is not an exclusive "or". That is, the OR in Condition D includes Conditions in which:

a. One or more required support or supported features, or both, are inoperable. . .; or

b. A Condition in which the turbine-driven auxiliary feedwater pump is inoperable; or

c. Both (a) and (b) above.}

{VS-BWR/4:
Condition D is one required DG inoperable AND one or more required support or supported features, or both, inoperable that are associated with the OPERABLE DGs, or with an opposite OPERABLE DC power subsystem, or both.}

{VS-BWR/6:
Condition D is one DG inoperable AND one or more required support or supported features, or both, inoperable that are associated the OPERABLE DGs, or with opposite OPERABLE DC power subsystems, or both.}

Condition D is a companion Condition to Condition C. That is, it is not possible to be in Condition D without also being in Condition C.

The rationale behind Condition D comes from GDC 33, 34, 35, 38, and 41. They state that,

"Suitable redundancy in components and features, and suitable interconnections, leakage detection, isolation, and containment capabilities shall be

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BASES (continued)

ACTIONS (continued)

provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure."

If, as per the GDC, we assume that all offsite power is not available, then Condition D represents a loss of function for the feature that is inoperable in the other {VS-BW.CE,W,BWR/4: [division] that has an OPERABLE DG or in the opposite OPERABLE DC power subsystem, or both.} {VS-BWR/6: [divisions] that have OPERABLE DGs or in opposite OPERABLE DC power subsystems, or both.}

Definition of DX: The allowable time for continued plant operation in Condition D is DX hours. DX is determined as follows. Consult the TS for the required feature that is inoperable. Define DXi as the Completion Time that the inoperable required feature TS allows for a complete loss of all required feature function. If no loss of function is allowed (e.g., if upon the loss of required feature function a shutdown is required), then assign DXi = 0 hours.

For each required feature that is inoperable, there will be a DXi. DX is then defined as the minimum of all the DXi; however, if DX is found to be less than 2 hours, DX is reset to 2 hours. If DX is found to be greater than 72 hours, then DX is 72 hours.

DX as defined above is acceptable because it minimizes risk while allowing time for restoration before subjecting the plant to transients associated with shutdown. (The above addresses the potential for loss of function under certain Conditions postulated in the design basis. In the event of an actual loss of function, the TS covering that loss of function will control the Completion Time.)

The specific list of "required support and supported features" encompassed by Condition D is provided in Reference 5. Required features are those that are designed with functionally redundant safety-related [divisions]. If a plant has a required feature that has no functionally redundant counterpart, that feature may not be required to be included. This is unlikely, however, since single-

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BASES (continued)

ACTIONS (continued)

failure considerations usually require functional redundancy of safety features. Since the Completion Time allowance for this Required Action is limited to 72 hours, those systems with allowed Completion Times ≥ 72 hours for complete loss of function are not included as required features to be checked.

{VS-BW,CE,W:
Auxiliary feedwater is provided by a [50%]-capacity motor-driven feedwater pump in [Division 1], a [50%]-capacity motor-driven feedwater pump in [Division 2], and a [100%]-capacity turbine-driven feedwater pump. Therefore, assuming that all offsite power is not available (as per the GDC), Condition D reduces the 72-hour Completion Time to DX hours for the case in which auxiliary feedwater function has been reduced to only [50%] of capacity or less.}

{VS-BW,CE,W:
The turbine-driven auxiliary feedwater pump is not included with the "one or more required support or supported features, or both, inoperable that are associated with the other [division] that has an OPERABLE DG" because the feedwater pump is steam driven (as opposed to motor driven), and thus is not "associated" with either [division] of the AC electrical power sources.}

{VS-BW,CE,W:
The Note for Required Action D.2.2 states, "Required Action D.2.2 is only required in MODES 1, 2, and 3, and in MODE 4 when auxiliary feedwater is being used for plant shutdown and startup." This Note is consistent with the Applicability requirements of Specification 3.7.4, "Auxiliary Feedwater System." When the pressure is < [715 psig] the turbine-driven auxiliary feedwater pump need not be capable of meeting the SR limits of SR 3.7.4.2 on developed head to satisfy the OPERABILITY requirements of Required Action D.2.2. The pump must be capable of coming up to speed and delivering flow, however. Furthermore, the licensee shall verify that the pump passed its last SR 3.7.4.2.}

Operation may continue in Condition D for a period that should not exceed DX hours. In this Condition, the remaining OPERABLE DG[s] and offsite circuits are adequate

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BASES (continued)

ACTIONS (continued)

to supply electrical power to the onsite Class 1E Distribution System. The DX-hour limit takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Thus, on a component basis, we may have lost single-failure protection for the required feature's function; however, we have not lost function. Similarly, we take into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period. If the Required Actions of Condition D and the associated Completion Times are not met, a controlled shutdown must be performed per Required Action J.1 and Required Action J.2.

E.1 and E.2

Condition E is one required offsite circuit inoperable AND one required DG inoperable. The Required Action is to either restore all required offsite circuits to OPERABLE status within a Completion Time of 12 hours OR restore all required DGs to OPERABLE status within a Completion Time of 12 hours. Condition E has been modified by a Note to indicate that when Condition E is entered with no AC source to one [division], LCO 3.8.7 must be immediately entered. Pursuant to the definition of OPERABILITY, this action should have already taken place; however, it is noted here to indicate that the Completion Time for Condition E under this situation is governed by the Completion Time of Required Action A.1 of LCO 3.8.7.

Per Regulatory Guide 1.93, "Availability of Elect . Power Sources" (Ref. 4), operation may continue in Condition E for a period that should not exceed 12 hours. The alternative Completion Time is for the situation in which Condition E was entered with no AC power to one [division], and the Completion Time to restore all required offsite circuits or DGs is then governed by LCO 3.8.7.

In Condition E, 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 F (loss of both required offsite circuits). This difference in reliability is offset by the

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BASES (continued)

ACTIONS (continued)

susceptibility of this power system configuration to a single bus or switching failure. The 12-hour or the alternate Completion Time limit takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period. If Required Action E.1 and Required Action E.2 and their associated Completion Times are not met, a controlled shutdown must be performed per Required Action J.1 and Required Action J.2.

F.1

Condition F is two required offsite circuits inoperable. Required Action F.1 is to restore at least
{VS-BW,CE,W,BWR/4: [one]}
{VS-BWR/6: two} required offsite
{VS-BW,CE,W,BWR/4: circuit[s]}
{VS-BWR/6: circuits} to OPERABLE status.
The intent of this Required Action is to restore either all required offsite circuits, or all but one required offsite circuit, to OPERABLE status within a Completion Time of 24 hours.

Per Regulatory Guide 1.93, "Availability of Electric Power Sources" (Ref. 4), operation may continue in Condition F for a period that should not exceed 24 hours. This degradation level 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 source have not been degraded. This degradation level 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 degradation level:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

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BASES (continued)

ACTIONS (continued)

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 design basis transient or accident. 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 limit provides a period of time to effect restoration of all or all but one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

Per Reference 4, with the available offsite AC source two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation may continue for a total time that should not exceed 72 hours (consistent with the loss of one AC source).

If no offsite source is restored within the first 24-hour period of continued operation, a controlled shutdown must be performed per Required Action J.1 and Required Action J.2.

G.1

Condition G is two required DGs inoperable. Required Action G.1 is to restore at least {VS-BW,CE,W,BWR/4: [one]} {VS-BWR/6: two} required diesel {VS-BW,CE,W,BWR/4: generator[s]} {VS-BWR/6: generators} to OPERABLE status.

The intent of this Required Action is to restore either all required DGs, or all but one required DG, to OPERABLE status within a Completion Time of 2 hours.

With two 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

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BASES (continued)

ACTIONS (continued)

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.

Per Reference 4, with both DGs inoperable, operation may continue for a period that should not exceed 2 hours. If both DGs are restored within 2 hours, unrestricted operation may continue. If only one DG is restored within these 2 hours, operation may continue for a total time that should not exceed 72 hours (consistent with the loss of one AC source). If no DG is restored within the first 2 hours of continued operation, a controlled shutdown must be performed per Required Action J.1 and Required Action J.2.

H.1

Condition H is three required AC sources inoperable. The Required Action is to enter LCO 3.0.3 immediately.

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 Surveil will cause a loss of function. Therefore, no additional time is justified for continued operation. The plant should be brought promptly to a controlled shutdown as required by LCO 3.0.3. During the shutdown process, the AC electrical power system should be critically monitored, and necessary actions taken, such as cross-connecting a supply to a load, if required, to ensure a safe shutdown.

I.1

Condition I is one required [automatic load sequencer] inoperable. The Required Action is to restore all required [automatic load sequencers] to OPERABLE status within the Completion Time of [2] hours [for Divisions 1 and 2].

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BASES (continued)

ACTIONS (continued)

{VS-BWR/6: If the sequencer is associated with [Division 3], then the Completion Time is [2 hours].}

{VS-BWR/6: [The [2-hour] Completion Time for an inoperable [Division 3] [automatic sequencer] is plant specific. Items to be considered in specifying this Completion Time for a given facility include:

a. The safety function of [Division 3]. If [Division 3] supports only the HPCS function, then there may not even be a [Division 3] [automatic sequencer] because there is only one large load to be connected to the [Division 3] ESF bus. If other ESF functions are supported by [Division 3], then the Completion Time for an inoperable [Division 3] [automatic sequencer] shall be [2 hours]; and

b. The safety function of the [Division 3] [automatic sequencer]:

   1. What is its role in mitigating a DBA?

   2. Does the [Division 3] [automatic sequencer] function as a support system to the [Division 3] DG, [Division 3] offsite circuit, or both? What ESF functions does it support?

   3. What is the role of the [Division 3] [automatic sequencer] in mitigating an SBO?}

Condition I corresponds to the sequencer(s) for [one ESF bus] being inoperable. If the sequencer(s) to [more than one ESF bus] are inoperable, enter LCO 3.0.3.

The sequencer(s) is (are) 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's sequencer] affects every major ESF system in the [division]. The [2]-hour Completion Time for [Divisions 1 and 2] {VS-BWR/6: and [2-hour] Completion Time for [Division 3]} provides a period of time to correct the problem commensurate with the importance of maintaining sequencer OPERABILITY. This time period also ensures that

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BASES (continued)

ACTIONS (continued)

the probability of an accident (requiring sequencer OPERABILITY) occurring during periods where the sequencer is inoperable is minimal.

[For plants that can show that the sequencer's role is less vital, a longer Completion Time may be appropriate. For example, if the ESF loads are block-loaded onto the offsite circuit so that no sequencer operation is required, then it may be possible to show that the sequencer is solely linked to DG OPERABILITY. In such a case, a Completion Time of [72 hours] may be appropriate.]

When a sequencer is inoperable, the associated [ESF bus] is declared inoperable, and LCO 3.8.7 is immediately entered. In LCO 3.8.7 it is determined whether the loss of functional capability exists by verifying whether one or more support or supported features, or both, are inoperable that are associated with the other ESF buses.

J.1 and J.2

The plant must be placed in a MODE in which the LCO does not apply if the inoperable AC electrical power sources and sequencers cannot be restored to OPERABLE status within the associated Completion Time. This is done by placing the plant in at least MODE 3 within {VS-BW,CE,W: 6 hours} {VS-GE: 12 hours} and in {VS-BW,CE,W: MODE 5} {VS-GE: MODE 4} within 36 hours. The allowed Completion Times are reasonable, based on operating experience related to the amount of time required to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The AC source are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with GDC 18 (Ref. 6). 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, "Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Power Plants" (Ref. 2); Regulatory Guide 1.108, "Periodic Testing of DG Units Used as Onsite Electric Power Systems at Nuclear Power Plants" (Ref. 7); and Regulatory Guide 1.137, "Fuel Oil Systems for Standby DGs" (Ref. 8), as addressed in the FSAR.

SR 3.8.1.1

This SR is required only when in Condition A, "One offsite circuit inoperable." Upon the inoperability of an offsite circuit, any remaining required offsite circuits that are OPERABLE must be checked for OPERABILITY within 1 hour of entering Condition A and once per 8 hours thereafter. If additional offsite circuits are found inoperable, they must be declared inoperable, and the corresponding Conditions of LCO 3.8.1 must be entered.

The requirement to perform SR 3.8.1.1 continues until LCO 3.8.1 is met, or until the plant is put in a MODE of operation outside of the Applicability of LCO 3.8.1.

This SR assures proper circuit continuity for the offsite AC power supply to the onsite distribution network and availability of offsite AC power. The breaker alignment verifies that each breaker is in its correct position to ensure distribution buses and loads are connected to their preferred power source. The check on devices that provide the separation and independence assures that protective relaying and interrupting devices are OPERABLE so that circuit independence can be maintained.

This Surveillance Frequency is justified based on the necessity to maintain a reliable AC electrical power system. The Frequency of 1 hour and once per 8 hours thereafter takes into account the time required to perform the Surveillance and the difficulty in completion. This is balanced against the desirability of having accurate and reliable information about remaining sources of offsite power upon the inoperability of one of the other offsite sources. Also, these Frequencies take into account the capacity, capability, redundancy, and diversity of the AC sources; other indications available in the control room, including alarms, to alert the operator to AC sources malfunctions; and the low probability of a DBA occurring during this period.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

It is recognized that an operator could choose not to perform SR 3.8.1.1 within 1 hour and once per 8 hours thereafter. Instead the operator could simply declare the second offsite circuit inoperable and accept a shorter Completion Time. While such action would be within the strict legal interpretation of the TS, it would not normally be prudent. In general, the operator should welcome the latest information on the condition of the plant. Furthermore, by failing to perform the SR on the second circuit, information on common cause failure may go undiscovered.

SR 3.8.1.2

This SR is required only when in Condition C, one DG inoperable. Upon the inoperability of a DG, any required offsite circuits that are OPERABLE must be checked for OPERABILITY within 1 hour of entering Condition C and once per 8 hours thereafter. If offsite circuit(s) are found inoperable, they must be declared inoperable, and the corresponding Conditions of LCO 3.8.1 must be entered.

The requirement to perform SR 3.8.1.2 continues until LCO 3.8.1 is met, or until the plant is put in a MODE of operation outside of the Applicability of LCO 3.8.1.

This SR assures 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 distribution buses and loads are connected to their preferred power source. The check on devices that provide the separation and independence assures that protective relaying and interrupting devices are OPERABLE so that circuit independence can be maintained.

This Surveillance Frequency is justified based on the necessity to maintain a reliable AC electrical power system. The Frequency of 1 hour and once per 8 eight hours thereafter takes into account the time required to perform the Surveillance and the difficulty in completion. This is balanced against the desirability of having accurate and reliable information about remaining sources of offsite electrical power upon the inoperability of one of the other offsite sources. Also these Frequencies take into account

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

the capacity, capability, redundancy and diversity of the AC sources; other indications available in the control room, including alarms, to alert the operators to AC sources malfunctions; and the low probability of a DBA occurring during this period.

It is recognized that an operator could choose not to perform SR 3.8.1.2 within 1 hour and once per 8 hours thereafter. Instead the operator could simply declare the offsite circuit inoperable and accept a shorter Completion Time. While such action would be within the strict legal interpretation of the TS, it would not normally be prudent. In general, the operator should welcome the latest information on the condition of the plant. Furthermore, by failing to perform the SR on the offsite circuit(s), information on common cause failure may go undiscovered.

SR 3.8.1.3

This SR is only required when in Condition C, one DG inoperable. Each and every required DG inoperability must be evaluated for common cause failure potential by performance of SR 3.8.1.3, regardless of when the DG is returned to OPERABLE status. If additional DGs are found inoperable, they must be declared inoperable, and the corresponding Conditions of LCO 3.8.1 must be entered.

The purpose of this SR is to determine absence of common cause for the DG inoperability for any remaining required DGs that are OPERABLE. This can be done either by analysis and reasoning (item A.1 of SR 3.8.1.3) or by starting the DG(s) that are OPERABLE (item B.1 of SR 3.8.1.3).

This Surveillance Frequency is justified based on the necessity to maintain a reliable AC electrical power system. The Frequency of once within [8] hours of entering Condition C takes into account the time required to perform the Surveillance and the difficulty in completion. This is balanced against the desirability of having accurate and reliable information about remaining sources of onsite electrical power upon the inoperability of one of the other onsite sources. Also these Frequencies take into account the capacity, capability, redundancy, and diversity of the AC sources; other indications available in the control room,

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

to alert the operators to AC sources malfunctions; and the low probability of a DBA occurring during this period.

SR 3.8.1.4

This SR assures 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 distribution buses and loads are connected to their preferred power source. The check on devices that provide the separation and independence assures that protective relaying and interrupting devices are OPERABLE so that circuit independence can be maintained. The 7-day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and its status is displayed in the control room.

SR 3.8.1.5 and SR 3.8.1.17

These SRs help to ensure the availability of the standby electrical power supply to mitigate design basis transients and accidents and maintain the plant 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 have been modified by a Note to indicate that all DG starts for these Surveillances may be preceded by an engine prelubricating period in accordance with vendor recommendations. For the purposes of this testing, the DGs shall be started from standby conditions.

Standby conditions for a [Division 1 or 2] DG means the diesel engine coolant and oil are being continuously circulated and temperature maintained consistent with manufacturer recommendations.
{VS-BWR/6: Standby conditions for [Division 3] DG means the lubricating oil is heated and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by the lubricating oil and circulates through the system by natural circulation.}

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

All engine starts for SR 3.8.1.5 may be preceded by warmup procedures as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine are minimized. This is the intent of Note 3 of SR 3.8.1.5.

SR 3.8.1.5 has been modified by a fourth Note, Note 4, requiring the performance of SR 3.8.1.6 immediately after SR 3.8.1.5. The exceptions (a) and (b) are for cases in which less than a full complement of AC sources, may be available. Therefore, the performance of SR 3.8.1.6 is not required because it requires the paralleling of two of the remaining AC sources, which may compromise the AC source independence.

SR 3.8.1.17 requires that, on a 184-day Frequency, the DG start from standby conditions and achieve required voltage and frequency within 10 seconds. The 10-second requirement supports the assump[illegible] design basis LOCA analysis (Ref. 9). The 10-second start requirement may not be applicable to SR 3.8.1.5 (see Note 3 of SR 3.8.1.5), which is usually performed on a 31-day Frequency. Since SR 3.8.1.17 does require a 10-second start, it is more restrictive than SR 3.8.1.5, and it may be performed in lieu of SR 3.8.1.5. This is the intent of Note 1 of SR 3.8.1.5. The normal 31-day Frequency for SR 3.8.1.5 (see DG test schedule, Table 3.8.1-1) is consistent with Regulatory Guide 1.9 (Ref. 2). The 184-day Frequency for SR 3.8.1.17 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 10). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.17 has been modified by a second Note, Note 2, which requires, following the completion of SR 3.8.1.17, the performance of SR 3.8.1.6. An exception is when SR 3.8.1.17 is required by SR 3.8.2.1. In this situation, less than a full complement of AC sources may be available. Therefore, the performance of SR 3.8.1.6 is not required because it requires the paralleling of two of the remaining AC sources, which may compromise the AC source independence.

<u>SR 3.8.1.6</u>

This Surveillance demonstrates that the DGs are capable of synchronizing and accepting greater than or equal to the

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

equivalent of the maximum expected accident loads. A third Note to this SR, Note 3, indicates that this Surveillance should only be conducted on one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. A minimum run time of 60 minutes is required to stabilize engine temperatures. Actual run time should be in accordance with vendor recommendations with regard to good operating practice and should be sufficient to ensure that cooling and lubrication are adequate for extended periods of operation, while minimizing the time that the DG is connected to the offsite source.

In order to assure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing shall be performed using a power factor in the range: [0.8] ≤ power factor ≤ [0.9]. This power factor range shall be chosen to be representative of the actual design basis inductive loading that the DG would experience. Alternatively, it may be conservatively chosen as a range that contains power factors that are numerically smaller than the power factors that are representative of the actual design inductive loading.

Note 1 modifies this Surveillance to indicate 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 (Ref. 10).

Note 2 modifies this Surveillance by stating that momentary transients because of changing bus loads do not invalidate this test. 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 (see DG test schedule, Table 3.8.1-1) is consistent with Regulatory Guide 1.9 (Ref. 2).

SR 3.8.1.7

This Surveillance verifies that, without the aid of the refill compressor, sufficient air-start capacity for each DG

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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. If the pressure is less than the value specified in this SR, the DG shall be declared inoperable. The five-start-cycles requirement is intended to provide redundancy for the DG start capability in the event that the hot DG does not start on the first attempt.

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 3.8.1.8

This SR provides verification that each DG day [and engine-mounted fuel] tank contains enough fuel oil, measured from the low-level alarm setpoint, to operate the DG for at least 1 hour at full load. If the day [and engine-mounted fuel] tank level is less than the required limit, the DG is inoperable.

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

SR 3.8.1.9

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 facility in a safe shutdown condition and to bring in replenishment fuel from an offsite location. If the storage tank level is less than the required limit, the DG is inoperable.

The 31-day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low-level alarms are

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

provided and facility operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.10

This Surveillance ensures that sufficient lubricating oil inventory is available to support at least 7 days of full-load operation for each DG. The [500]-gal requirement is based on the DG manufacturer's consumption values for the run time of the diesel. Implicit in this SR is the requirement to verify the capability to transfer the lubricating-oil from its storage location to the DG. If it can be demonstrated that the DG lubricating-oil sump can hold adequate inventory for 7 days of full-load operation without the level reaching a dangerous point, then the quantity or level of lubricating oil in the sump can be used in this SR. If the lubricating oil inventory is less than the limit, the DG is inoperable.

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

SR 3.8.1.11

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/operation. 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. The tests, limits, and applicable American Society for Testing Materials (ASTM) standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4054-[ ];

b. Verify in accordance with the tests specified in ASTM D975-[ ] that the sample has an absolute specific gravity at 60/60°F of $\geq$ 0.83° but $\leq$ 0.89° or an API gravity at 60°F of $\geq$ 27° but $\leq$ 39°, a kinematic viscosity at 40°C of $\geq$ 1.9 centistokes but $\leq$ 4.1 centistokes, and a flash point $\geq$ 125°F; and

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-[ ].

These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case shall the time between receipt of new fuel and conducting the tests exceed 31 days.

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

SR 3.8.1.12

Within 31 days following the initial new fuel-oil sample, this Surveillance is performed to establish that the other properties specified in Table 1 of ASTM D975-[ ] are met for new fuel oil when tested in accordance with ASTM D975-[ ], except that the analysis for sulfur may be performed in accordance with ASTM D1522-[ ] or ASTM D2622-[ ]. 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. For the same reason, should one or more of these properties not be within limits, there is no need to declare the DG inoperable. It is acceptable to continue operation for up to [31] days while measures are taken to ensure that the properties of the mixed fuel oil are within limits or that the fuel-oil properties are being restored to within limits. If after continued operation for [31] days the properties of the mixed fuel oil are still not within limits, the DG shall be declared inoperable.

SR 3.8.1.13

This Surveillance is an integral part of a comprehensive program to ensure the availability of high-quality fuel oil for the DGs at all times. By testing for particulate on a 31-day basis, information regarding the condition of stored fuel oil can be obtained and trended.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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. If particulate is removed from stored fuel oil by circulating the oil through filters (other than diesel engine filters), the fuel oil can be restored to acceptable condition and its storage life extended indefinitely. By obtaining and trending particulate data, it is possible to determine when stored-fuel-oil cleanup will be necessary. This is done before the maximum allowable particulate concentration is reached

Particulate concentrations should be determined in accordance with ASTM D2276-[ ], Method A. 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 lab testing in lieu of field testing. In the case(s) where 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 Surveillance takes into consideration fuel-oil degradation trends that indicate that particulate concentration is unlikely to change between Frequency intervals.

There is no quantitative data regarding the capability of diesel engines to operate for prolonged periods of time with fuel-oil particulate concentrations in excess of 10 mg/l. Therefore, if this limit is reached, the associated DG shall be declared inoperable. In practice, however, this should not present a problem since the concept behind this SR is to establish fuel-oil degradation trends, which will provide an alert to the need for corrective action prior to impacting on DG OPERABILITY.

<u>SR 3.8.1.14 and SR 3.8.1.15</u>

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

fuel-oil day [and engine-mounted] tanks and from storage tanks once every 31 days will eliminate the necessary environment for survival. This is the most effective means of controlling microbiological fouling. In addition, it will eliminate 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 will minimize fouling as well as provide data regarding the watertight integrity of the fuel-oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 8).

SR 3.8.1.16

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 the 7-day continuous operation of standby power sources. This Surveillance provides assurance that the fuel-oil transfer pump is OPERABLE, the fuel-oil piping system is intact, the fuel-delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE. The 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 Section XI code; 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 dur[illegible]r following DG testing. In such a case a 31-day Frequ[illegible]s appropriate. Since proper operation of fuel-transfer s[illegible]tems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs. Upon failure of this SR, the DG shall be declared inoperable immediately.

SR 3.8.1.17

See SR 3.8.1.5.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.18

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 plant 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 on the [18-month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR has been modified by a Note, Note 1, which states that the SR must not be performed in MODE 1 or 2. The reason for this is that during operation with the reactor critical, performance of this SR could potentially cause perturbations to the electrical distribution systems that could result in a challenge to continued steady-state operation and, as a result, to plant safety systems.

Note 2 has been included in this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.19

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 this facility, the largest single load for each DG and its horsepower rating is as follows:] As required by IEEE-308, 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. [For this facility, the SR 3.8.1.19

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

frequency ([63] Hz) for each DG and one of the two above criteria used to arrive at this number are as follows:]

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 2) recommendations for response during load sequence intervals. The [3] seconds specified is equal to 60% of a typical 5-second interval. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.19a corresponds to the maximum frequency excursion, while SR 3.8.1.19b and SR 3.8.1.19c are steady-state voltage and frequency values that the system must recover to following load rejection. The [18-month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 7).

In order to assure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing shall be performed using a power factor in the range: [0.8] ≤ power factor ≤ [0.9]. This power factor range shall be chosen to be representative of the actual design basis inductive loading that the DG would experience. Alternatively, it may be conservatively chosen as a range that contains power factors that are numerically smaller than the power factors that are representative of the actual design basis inductive loading. If the facility uses the actual single largest load to perform this test, then the power factor will be set by that load.

This SR has been modified by a Note, Note 1, which states that the SR must not be performed in MODE 1 or 2. The reason for this is that during operation with the reactor critical, performance of this SR could potentially cause perturbations to the electrical distribution systems that could result in a challenge to continued steady-state operation and, as a result, to plant safety systems.

Note 2 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

<u>SR 3.8.1.20</u>

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

may occur because of a system fault or inadvertent breaker tripping. This Surveillance verifies propc engine-generator load response under the simulated test conditions. This test will simulate the loss of the total connected load that the DG will experience following a full-load rejection and verify that the DG will not trip upon loss of the load. These acceptance criteria provide for DG damage protection. While the DG is not expected to experience this transient during an event and continue to be available, this response will assure 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 assure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing shall be performed using a power factor in the range: [0.8] $\leq$ power factor $\leq$ [0.9]. This power factor range shall be chosen to be representative of the actual design basis inductive loading that the DG would experience. Alternatively, it may be conservatively chosen as a range that contains power factors that are numerically smaller than the power factors that are representative of the actual design basis inductive loading.

This SR has been modified by a Note, Note 1, which states that the SR must not be performed in MODE 1 or 2. The reason for this is that during operation with the reactor critical, performance of this SR could potentially cause perturbation to the electrical distribution systems that could result in a challenge to continued steady-state operation.

The [18-month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 7) and is intended to be consistent with expected fuel-cycle lengths.

Note 2 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.21

As required by Regulatory Guide 1.108 (Ref. 7), 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 automatic start time of [10] seconds is derived from requirements of the accident analysis to respond to a design basis large-break LOCA. The minimum steady-state output voltage of [3744] V is [90%] of the nominal [4160 V] output voltage. This value allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% of 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 nameplate rating.

The specified maximum steady-state output voltage of 4576 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 will be no more than the maximum rated operating voltages.

The specified minimum and maximum steady-state output frequency of the DG is [58.8] Hz and [61.2] Hz respectively. This is equal to ± 2% of the 60 Hz nominal frequency and is derived form the recommendations given in Regulatory Guide 1.9 (Ref. 2) that the frequency should be restored to within 2% of nominal following a load sequence step. 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 Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 7), paragraph 2.a.(1), takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel-cycle lengths.

This SR has been modified by a Note, Note 1, stating that all DG starts may be preceded by prelubricating procedures as recommended by the manufacturer. The reason for this is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs shall be started from

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for [Division 1 and 2] DGs. {VS-BWR/6: For the [Division 3] DG, standby conditions means the lubricating oil is heated and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by the lubricating oil and circulates through the system by natural circulation}.

This SR has been modified by a second Note, Note 2, which states that the SR must not be performed in {VS-BW,CE,W: MODE 1, 2, 3, or 4} {VS-GE: MODE 1, 2, or 3}. The reason for this is that performing the SR would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

Note 3 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.22

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 ≥ [5] minutes. The [5]-minute period provides sufficient time to demonstrate stability. SR 3.8.1.22d and SR 3.8.1.22e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on a ESF signal without loss of offsite power. The bases for the time, voltage, and frequency tolerances specified in this Surveillance are discussed under SR 3.8.1.21, above.

This SR has been modified by a Note, Note 1, which states that all DG starts may be preceded by prelubricating procedures as recommended by the manufacturer. The reason for this is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs shall be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for [Division 1 and 2] DGs. {VS-BWR-6: For the [Division 3] DG, standby conditions means the lubricating oil is heated and continuously circulated

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

through a portion of the system as recommended by the vendor. Engine jacket water is heated by the lubricating oil and circulates through the system by means of natural circulation).

This SR has been modified by a second Note, Note 2, which states that the SR must not be performed in MODE 1 or 2. The reason for this is that during operation with the reactor critical, performance of this SR could potentially cause perturbations to the electrical distribution systems that could result in a challenge to continued steady-state operation and, as a result, to plant safety systems.

Note 3 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

The Frequency of [18 months] takes into consideration plant 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 on the [18-month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.8.1.23

This Surveillance demonstrates that DG non-critical 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, and low lubricating oil pressure) trip the DG to avert substantial damage to the DG unit. The non-critical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This 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 plant 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

performed on the [18-month] Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR has been modified by a Note, Note 1, which states that the SR must not be performed in {VS-BW,CE,W: MODE 1, 2, 3, or 4} {VS-GE: MODE 1, 2, or 3}. The reason for this is that performing the SR would remove a required DG from service.

Note 2 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.24

Regulatory Guide 1.108 (Ref. 7), 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, 22 hours of which is at a load equivalent to the continuous rating of the DG and 2 hours of which is at a load equivalent to the 2-hour 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 3.8.1.5, and for gradual loading, discussed in SR 3.8.1.6, are applicable to this SR.

In order to assure that the DG is tested under load conditions that are as close to design conditions as possible, testing shall be performed using a power factor in the range: [0.8] ≤ power factor ≤ [0.9]. This power factor range shall be chosen to be representative of the actual design basis inductive loading that the DG would experience. Alternatively, it may be conservatively chosen as a range that contains power factors that are numerically smaller than the power factors that are representative of the actual design basis inductive loading.

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance has been modified by a Note, Note 1, which states that momentary transients due to changing bus loads do not invalidate this test. 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.

This SR has been modified by a second Note, Note 2, which states that the SR must not be performed in MODE 1 or 2. The reason for this is that during operation with the reactor critical, performance of this SR could potentially cause perturbations to the electrical distribution systems that could result in a challenge to continued steady-state operation and, as a result, to plant safety systems.

Note 3 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.25

This Surveillance demonstrates that the diesel engine can restart from a hot condition 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 requirement that the diesel have operated for at least 2 hours at full-load conditions prior to performance of this Surveillance is based on manufacturer's recommendations for achieving hot conditions. The bases for the voltage and frequency tolerances are discussed in the Bases for SR 3.8.1.21.

The Surveillance demonstrates the DG capability to respond to accident signal while hot, such as subsequent to shutdown from normal Surveillances. The [18-month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 7), paragraph 2.a.(5).

In order to assure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing shall be performed using a power factor in the range: [0.8] ≤ power factor ≤ [0.9]. This power factor range shall be chosen to be representative of the actual design basis inductive loading that the DG would experience.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Alternatively, it may be conservatively chosen as a range that contains power factors that are numerically smaller than the power factors that are representative of the actual design basis inductive loading.

This SR has been modified by a Note, Note 1, which states that the SR shall be performed within 5 minutes of shutting down the DG after it has operated more than 2 hours at between [5450 and 5740] kW. This is to ensure that the test is performed with the diesel sufficiently hot.

This SR has been modified by a second Note, Note 2, which states that all DG starts may be preceded by prelubricating procedures as recommended by the manufacturers. The reason for this is to minimize wear and tear on the diesel during testing.

This Surveillance has been modified by a third Note, Note 3, which states that momentary transients due to changing bus loads do no invalidate this test. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

SR 3.6.1.26

As required by Regulatory Guide 1.108 (Ref.7), paragraph 2.a.(6), this Surveillance assures 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. 7), paragraph 2.1.(6), and takes into consideration plant conditions required to perform the Surveillance.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

This SR has been modified by a Note, Note 1, which states that the SR must not performed in {VS-BW,CE,W: MODE 1, 2, 3, OR 4} {VS-GE: MODE 1, 2, or 3}. The reason for this is that performing the SR would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

Note 2 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.27

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing. Interlocks to the LOCA sensing circuits cause the DG to automatically reset to ready-to-load operation if a LOCA actuation signal is received during operation in the test 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. 11), paragraph 6.2.6(2).

The [18-month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 7), paragraph 2.a.(8), takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel-cycle lengths.

This SR has been modified by a Note, Note 1, which states that the SR must not be performed in {VS-BW,CE,W: MODE 1, 2, 3, or 4} {VS-GE: MODE 1, 2 or 3}. The reason for this is that performing the SR would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

Note 2 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.28

As required by Regulatory Guide 1.108 (Ref. 7), paragraph 2.a.(2), each DG is required to demonstrate proper operation for the DBA loading sequence to ensure that voltage and frequency are maintained within the required limits. Under

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

accident conditions, prior to connecting the diesel generators to their appropriate bus, all loads are shed except load center feeders and those motor control centers that power Class 1E loads (referred to as "permanently connected" loads). Upon reaching rated voltage and frequency, the DGs are then connected to their respective bus. Loads are then 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 3 provides a summary of the automatic loading of ESF buses.

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

This SR has been modified by a Note, Note 1, which states that the SR must not be performed in {VS-BW,CE,W: MODE 1, 2, 3, or 4} {VS-GE: MODE 1, 2, or 3}. The reason for this is that performing the SR would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

Note 2 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.29

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). SR 3.8.1.29b and SR 3.8.1.29c ensure that permanently connected loads remain energized from the offsite electrical power system, and that emergency loads are energized [or auto-connected through the load sequencer] to the offsite electrical power system. Before the last [sequencer] load step, a loss of offsite power is simulated. It must then be shown that the AC

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

sources and sequencer reset themselves so that the powering of the loads can begin all over again, this time with the DG as the power source.

This SR has been modified by a Note, Note 1, which states that all DG starts may be preceded by prelubricating procedures as recommended by the manufacturer. The reason for this is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs shall be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for [Division 1 and 2] DGs. {VS-BWR/6: For the [Division 3] DG, standby conditions means the lubricating oil is heated and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by the lubricating circulation.}

This SR has been modified by a second Note, Note 2, which states that the SR must not be performed in {VS-BW,CE,W: MODE 1, 2, 3, or 4} {VS-GE: MODE 1, 2, or 3}. The reason for this is that performing the SR would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

Note 3 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

The Frequency of [36 months] alternated with SR 3.8.1.30 means that once within [18 months] either SR 3.8.1.29 or SR 3.8.1.30 is completed for each DG. Then once within the following [18 months] the other SR, SR 3.8.1.30 or SR 3.8.1.29, is completed for each DG. This Frequency takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel-cycle length of [18 months]. [For this facility, operating experience has demonstrated that the Frequency for this SR is adequate for the following reasons:]

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.30

In the event of 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 under SR 3.8.1.22 above, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal.

The Frequency of [36 months] alternated with SR 3.8.1.29 means that once within [18 months] either SR 3.8.1.29 or SR 3.8.1.30 is completed for each DG. Then once within the following [18 months] the other SR, SR 3.8.1.30 or SR 3.8.1.29, is completed for each DG. This Frequency takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel-cycle length of [18 months]. For this facility, operating experience has demonstrated that the Frequency for this SR is adequate for the following reasons:]

This SR has been modified by a Note, Note 1, which states that all DG starts may be preceded by prelubricating procedures as recommended by the manufacturer. The reason for this is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs shall be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for [Division 1 and 2] DGs. {VS-BWR/6: For the [Division 3] DG, standby condition means the lubricating oil is heated and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by the lubricating oil and circulates through the system by means of natural circulation}.

This SR has been modified by a second Note, Note 2, which states that the SE must not be performed in {VS-BW,CE,W: MODE 1, 2, 3, or 4} {VS-GE: MODE 1, 2, or 3}. The reason for this is that performing the SR would remove a required

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

Note 3 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.31

Draining of the fuel oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning are required at 10-year intervals by Regulatory Guide 1.137 (Ref. 8), paragraph 2.f. This Sr also requires the performance of the Section XI examinations of the tanks. To preclude the introduction of surfactants in the fuel system, the cleaning should be accomplished using sodium hypochlorite solutions, or their equivalent, rather than soap or detergents.

SR 3.8.1.32

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. 7), paragraph 2.b and Regulatory Guide 1.137 (Ref. 8), paragraph C.2.f.

This SF has been modified by a Note that all DG starts may be preceded by prelubricating procedures as recommended by the manufacturer. The reason for this is to minimize wear on the DG during testing. For the purpose of this testing, the DGs shall be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations. {VS-BWR/6: Standby conditions for [Division 3] DG means the lubricating oil is heated and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by the lubricating oil and circulates through the system by means of natural circulation.}

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

DG Test Schedule

The DG test schedule (Table 3.8.1-1) implements the recommendations of Revision 3 to Regulatory Guide 1.9 (Ref. 2). The purpose of this test schedule is to provide sufficiently timely that is test data to establish a confidence level associated with the goal to maintain DG reliability above 0.95 per demand.

Per Regulatory Guide 1.9, Revision 3, each DG unit should be tested at least once every 31 days. Whenever a DG has experienced four or more valid failures in the last 25 demands, the maximum time between tests is reduced to 7 days. Four failures in 25 demands 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, four failures in the last 25 demands 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.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, General Design Criterion 17, "Electric Power Systems."
2. Regulatory Guide 1.9, Rev. [ ], "Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," [date].
3. [Plant Name] FSAR, Tables [8.3-1 to 8.3-3], "[Title]."
4. Regulatory Guide 1.93, Rev. [ ], "Availability of Electric Power Sources," [date].
5. [List of equipment (required features) referred to in Conditions B and D].

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BASES (continued)

REFERENCES (continued)

6. Title 10, Code of Federal Regulations, Part 50, General Design Criterion 18, "inspection and Testing of Electric Power Systems."
7. Regulatory Guide 1.108, Rev. [ ], "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," [ ].
8. Regulatory Guide 1.137, Rev. [ ] "Fuel Oil Systems for Standby Diesel Generators," [date].
9. [[Plant Name] FSAR, Section [ ]], [This reference is to provide the assumptions of the design basis LOCA].
10. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
11. IEEE Standard 308-[ ], "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources—Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for Specification 3.8.1, "AC Sources—Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC and DC power sources and associated distribution systems during shutdown and refueling, as specified in the LCO, ensures that (Ref. 1):

a. The facility 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 an inadvertent draindown of the vessel or a fuel-handling accident.

Although in many cases the FSAR may only address bounding analyses that are typically for power operation, for other modes of operation, the GDC (Ref. 2), among other requirements, are still required to be met. As these GDC are not MODE specific, and as it is a function of the Technical Specifications (TS) to ensure that the plant 's operated within its design basis, with regard to AC sources, the requirements established in the TS must be consistent with the GDC related to electrical systems, as well as with other GDC related to safety-related systems, since the AC sources comprise a typical support system.

In general, when the plant is shut down the TS requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents assuming a single failure, because either:

a. Redundant and independent systems are required to be OPERABLE; or

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

b. Appropriate administrative measures are established and/or alternate backup systems that can provide functional redundant capability are required to be OPERABLE or put into operation in a period of time commensurate with the accident and the initial conditions considered.

This statement, in general, is reflected in the system LCOs for shutdown MODES of operation.

In addition to the postulated shutdown events directly addressed in the plant FSAR, it is necessary to consider evaluations of plant data that show that a large number of events can take place during shutdown. If not mitigated, some of these events can lead to core damage. Typically, the loss of decay-heat removal while there is substantial core decay heat poses a significant likelihood of a release due to a severe core damage accident.

To avoid the consequences of possible accidents during shutdown, different requirements are established according to the design of each plant. So, as far as residual heat removal (RHR) is concerned {VS-BW,CE,W: the OPERABILITY of the two RHR loops is required in MODES 5 and 6 when the reactor coolant loops are not filled (MODE 5) and when the Reactor Coolant System (RCS) water level above the top of the reactor vessel flange is less than 23 feet (MODE 6). See Specifications 3.4.8, "RCS Loops - MODE 5, Loops Not Filled," and {VS-W: 3.9.7, "Residual Heat Removal and Coolant Circulation—Low Water Level."} {VS-CE: 3.9.5, "Shutdown Cooling and Coolant Circulation—Low Water Level."} {VS-BW: 3.9.5, "Decay Heat Removal and Coolant Circulation - Low Water Level."}} {VS-GE: The OPERABILITY of the two Residual Heat Removal shutdown cooling subsystems is always required in MODE 4, and in MODE 5 when RCS water level above the top of the reactor vessel flange is less than 23 feet. See Specifications {VS-BWR/4: 3.4.8,} {VS-BWR/6: 3.4.9,} "Residual Heat Removal—Shutdown," and 3.9.8, "Residual Heat Removal—Low Water Level."} Therefore, in these conditions, [Division 1 and 2] AC sources are required to be OPERABLE as support systems.

Furthermore, by application of GDC 34, "Residual Heat Removal," and the design basis definition of <u>operability</u> (See AC Sources and Component OPERABILITY, Bases for

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Specification 3.8.1), it is clear that each RHR pump must be backed up by separate and independent onsite and offsite sources.

Thus, to meet the design basis definition of operability and GDC 34, four AC sources are required when two RHR pumps are required OPERABLE. As discussed above, however, each plant may have put in additional measures to help mitigate the potential consequences of an accident in these operating MODES. For those plants, Specification 3.8.2 is written such that three out of four AC sources will suffice.

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

LCO

LCO 3.8.2.a and LCO 3.8.2.b require that one offsite circuit and one diesel generator be OPERABLE (see Bases 3.8.1) and capable of supplying the onsite Class 1E power distribution subsystem of LCO 3.8.8.a. The intent is that all required non-redundant loads, as well as one required load from each required redundant pair of loads, be powered from the same safety [division] and that all required AC and DC sources, as well as the distribution subsystem itself, will be OPERABLE so that the AC and DC sources and the distribution subsystem will be capable of fully supporting the non-redundant loads.

When redundant counterpart loads (e.g., the second members of the pair) are required to be OPERABLE, LCO 3.8.2.c requires that they be powered by a third separate and independent, readily available AC source. Readily available means that the source can be made OPERABLE and put into operation, if necessary, within a time commensurate with the safety importance of the redundant loads.

{VS-BWR/6: LCO 3.8.2.d requires an offsite circuit to power the high pressure core spray (HPCS) system when it is required to be OPERABLE, or when other loads assigned to the HPCS system [division] are required to be OPERABLE, or both. The requirements set forth in this LCO may need to be restructured depending on the functions required to be accomplished during these modes of operation by the required loads assigned to [Division 3]. [For this facility, the

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BASES (continued)

LCO (continued)

functions associated with the required loads assigned to [Division 3] during these modes of operation are as follows:]

See the Bases of Specification 3.8.1 for additional information on AC source OPERABILITY and AC source support and supported systems.

LCO 3.8.2 specifies the minimum AC sources required to be OPERABLE in MODES {VS-BW,CE,W: 5 and 6} {VS-GE: 4 and 5} and any time when handling irradiated fuel {VS-GE:[or moving loads over irradiated fuel in the primary or secondary containment]}. It ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel-handling accidents, reactor vessel draindown).

As described in the previous section, "Applicable Safety Analyses," in the event of an accident during shutdown, the TS are designed to maintain the plant in a condition such that, even with a single failure, the plant will not be in immediate difficulty. In some cases, this is accomplished by requiring completely redundant and independent systems to be OPERABLE. In other cases, if justified based on a single plant design, administrative measures may be sufficient to relax the single-failure criterion. Also, an alternative backup system that provides the same functional capability may be substituted provided the backup system is OPERABLE or can be made OPERABLE in sufficient time to mitigate the consequences of an accident during shutdown. When required to be OPERABLE, systems are reliable only if their support requirements are also met. The AC sources comprise a typical support system.

APPLICABILITY

The AC sources required to be OPERABLE in MODES {VS-BW,CE,W: 5 and 6} {VS-GE: 4 and 5} and also any time when handling irradiated fuel {VS-GE: [or moving loads over irradiated fuel in the primary or secondary containment]} provide assurance that:

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BASES (continued)

APPLICABILITY (continued)

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

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

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

AC power requirements for {VS-BW,CE,W: MODES 1, 2, 3, and 4} {VS-GE: MODES 1, 2, and 3} are covered in Specification 3.8.1, "AC Sources—Operating."

ACTIONS

A.1, A.2, A.3, A.4, A.5, and A.6

With one or more of the required AC electrical power sources inoperable, some equipment is not receiving the minimum support it needs. It is, therefore, required to suspend CORE ALTERATIONS, handling of irradiated fuel, {VS-GE: moving of loads over irradiated fuel,} any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions will preclude the occurrence of actions that could potentially initiate the 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's 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

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BASES (continued)

ACTIONS (continued)

should be completed as quickly as possible in order to minimize the time the unit's safety systems may be without power.

Required Action A.6 verifies that the Required Actions have been initiated for those supported systems declared inoperable as a result of the total loss of power to a power distribution subsystem within the same Completion Time as that specified for Required Action A.5.

This Required Action has been modified by a Note to clarify that Required Action A.6 needs to be executed only when there are no AC power sources to one or more [divisions] of the onsite Class 1E Power Distribution System.

Required Action A.6 ensures that those identified Required Actions associated with supported systems affected by the total loss of power to a [division] of AC and DC power distribution subsystem have been initiated by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition A of this LCO.]

[For this facility, the identified support systems' Required Actions are as follows:]

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 lists 16 SRs from LCO 3.8.1 that are required to be met. Therefore, see the corresponding Bases for Specification 3.8.1 for a discussion of each SR.

REFERENCES

1. [Unit name] FSAR, Section [ ], "[Title]."

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 DC Sources—Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety-related equipment. 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 requirements of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

{VS-BW,CE,W,BWR/4: The [250/125] Vdc electrical power system consists of two independent and redundant safety-related Class 1E DC electrical power subsystems ([Division 1 and 2]).} {VS-BWR/6: The [250/125] Vdc electrical power system consists of three independent Class 1E DC electrical power subsystems ([Divisions 1, 2, and 3]).} Each subsystem consists of [two] battery banks [(each bank [50%] capacity)], associated battery charger(s), ([one] per bank), and all the associated control equipment and interconnecting cabling. [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 [250/125] Vdc load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

Each of the [Division 1 and 2] electrical power subsystems provides the control power for its associated Class 1E AC-power-load group, [4.16] kV switchgear, and [480] V load centers. Also, these DC subsystems provide DC electrical power to the inverters, which in turn power the AC vital buses. {VS-BWR/6: The [Division 3] DC electrical power

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BASES (continued)

BACKGROUND (continued)

subsystem provides DC motive and control power as required for the High Pressure Core Spray System diesel generator (DG) set control and protection, and all [Division 3]-related control.]}

The DC-power distribution system is described in more detail in Bases for Specifications 3.8.7, "Distribution System—Operating," and 3.8.8, "Distribution System—Shutdown."

In the event of loss of all unit AC power, which is beyond the design bases, the DC system is the only electrical power source available to monitor critical plant parameters and operate selected equipment.

Each battery bank of the [Division 1 and 2] DC electrical power subsystem consists of [120] lead-[calcium] cells with a continuous discharge rating of [1650] Ah for [8] hours to [210] Vs at [77]°F. Plant battery operating voltage is [250/125] Vs, and 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 (Ref. 4). Capacity is adequate for loss-of-coolant accident (LOCA) conditions or any other emergency shutdown.

{VS-BWR/6: The [Division 3] DC electrical power subsystem consists of a [125] V, [60]-cell lead-calcium battery with a continuous discharge of [1000] Ah for [8] hours to [105] V at [77]°F; the battery has adequate storage to carry the required load continuously for at least [2] hours and to perform [three] complete cycles of intermittent loads (Ref. 4). Capacity is adequate for LOCA conditions or any other emergency shutdown.}

The battery chargers of [Division 1 and 2] DC electrical power subsystems are rated at [300] amps with 0.5% voltage regulation with an AC-supplied variation of [480 V ± 15%] in voltage and [60 Hz ± 5%] in frequency (Ref. 4).

{VS-BWR/6: The battery charger for [Division 3] DC electrical power subsystem is rated at [150] amps with 0.5% voltage regulation with an AC-supplied variation of [480 V ± 15%] in voltage and [60 Hz ± 5%] in frequency (Ref. 4).}

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BASES (continued)

BACKGROUND (continued)

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

Battery rooms are continuously ventilated in order to prevent accumulation of hydrogen and to maintain design temperature. The ventilation system limits the hydrogen accumulation to less than [1]% of the total of battery room volume (Ref. 4). The threshold of ignition is 4% and maximum hydrogen generation occurs during overcharging.

The batteries for [Division 1 and 2] DC electrical power subsystem 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 bank (Ref. 4). The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

{VS-BWR/6: The battery for [Division 3] DC electrical power subsystem 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 bank (Ref. 4).}

Each battery charger of [Division 1 and 2] 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 has sufficient capacity to restore the battery bank from the

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BASES (continued)

BACKGROUND (continued)

design minimum charge to its fully charged state within 24 hours while supplying normal steady-state loads (Ref. 4).

{VS-BWR/6: The battery charger of [Division 3] DC electrical power subsystem has sufficient capacity to restore the battery bank from the design minimum charge to its fully charged state in [8] hours while supplying normal steady-state loads (Ref. 4)].}

APPLICABLE SAFETY ANALYSES

The initial conditions of design basis transient and accident analyses in the FSAR, [Chapter 6, "Engineered Safety Features"], and [Chapter 15, "Accident Analyses"], 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 sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one [division] of the onsite power or offsite AC sources, DC sources, and associated distribution systems 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.

DC Sources—Operating satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

As described in the Background section, each [divisional] DC electrical power subsystem consists of [two] battery bank(s), associated battery charger(s) and the corresponding control equipment and interconnecting cabling within the [division].

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BASES (continued)

LCO (continued)

All DC electrical power subsystems are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated Design Basis Accident (DBA). Loss of any [divisional] DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

A DC electrical power subsystem is OPERABLE provided:

a. All of its required battery bank(s) and battery charger(s) are connected to their associated DC bus(es) and are operating; and

b. All of its required battery bank(s) and battery charger(s) are OPERABLE.

Furthermore, for DC subsystems to be OPERABLE, they must be capable of performing their intended functions, have all support systems OPERABLE, and have successfully completed all SRs.

[For this facility, an OPERABLE [divisional] DC electrical power subsystem consists of the following:]

[For this facility, the following support systems are required OPERABLE to ensure [divisional] DC electrical power subsystem OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not declare DC electrical power subsystems inoperable and their justification are as follows:]

[For this facility, the supported systems affected by the inoperability of a DC electrical power subsystem and the justification for whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES {VS-BW,CE,W: 1, 2, 3, and 4}{VS-GE: 1, 2, and 3} to ensure safe plant operation and to ensure that:

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BASES (continued)

APPLICABILITY (continued)

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

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

DC electrical power requirements for MODES {VS-BW,CE,W: 5 and 6} {VS-GE: 4 and 5} are addressed in the Bases for Specification 3.8.4, "DC Sources—Shutdown."

ACTIONS

A.1 and A.2

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 {VS-BW,CE,W,BWR/4: subsystem has} {VS-BWR/6: subsystems have} the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst-case single failure would, however, result in {VS-BW,CE,W,BWR/4: the complete loss of the [250/125] Vdc electrical power system} {VS-BWR/6: only one DC electrical power subsystem being OPERABLE} 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. 6) and reflects a reasonable time to assess plant status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, prepare to effect an orderly and safe plant shutdown. {VS-BWR/6: However, if the inoperable DC electrical power subsystem is associated with [Division 3], then continued operation for up to a [2-hour] Completion Time is plant specific and is meant to be the most limiting Completion Time for all systems that a [Division 3] DC electrical power subsystem supports; furthermore, the number chosen for the [2-hour] Completion Time is not to exceed 8 hours if more than two systems are made inoperable because of the [Division 3] DC electrical power subsystem inoperability.

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BASES (continued)

ACTIONS (continued)

For example, if the [Division 3] batteries support only the [Division 3] DG, then a Completion Time of [72 hours] would be appropriate, consistent with the Completion Time for an inoperable [Division 3] DG.

If the [Division 3] batteries support both the [Division 3] DG and the [Division 3] offsite circuit, then the Completion Time will be governed by Condition E of Specification 3.8.1.

If the [Division 3] batteries support even more items, such as a [Division 3] sequencer or other [Division 1 and 2] ESF functions, then a [2-hour] Completion Time is appropriate.}

Required Action A.2 verifies that the Required Actions for those supported systems declared inoperable because of the inoperability of one [division] DC electrical power subsystem have been initiated and within the same Completion Time as that of Required Action A.1.

Required Action A.2 ensures that those identified Required Actions associated with supported systems affected by the inoperability of the [division] DC electrical power subsystem have been initiated. This can be accomplished by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition A of this LCO.]

[For this facility, the identified supported system Required Actions are as follows:]

B.1

With two {VS-BWR/6: or more} required [divisions of] DC electrical power subsystems inoperable, the plant is in a condition outside the accident analysis as discussed in A.1, above. Therefore, LCO 3.0.3 must be entered immediately.

C.1

With one [division] DC electrical power subsystem inoperable AND one or more required support or supported features, or both, inoperable associated with the OPERABLE [division] of DC electrical power subsystems, or with opposite OPERABLE AC and DC electrical power distribution subsystems, or both,

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BASES (continued)

ACTIONS (continued)

there is a loss of functional capability and LCO 3.0.3 must be immediately entered. However, if the LCOs for the support or supported feature, or for both, take into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

D.1 and D.2

If the DC electrical power subsystem cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within {VS-BW,CE,W: 6} {VS-GE: 12} hours and in MODE {VS-BW,CE,W: 5} {VS-GE: 4} within 36 hours. The Completion Times are reasonable, based on operating experience related to the amount of time required to reach the required MODES from full power in an orderly manner and without challenging plant systems. The Completion Time to bring the unit to MODE {VS-BW,CE,W: 5} {VS-GE: 4} is consistent with the time required in Regulatory Guide 1.93 (Ref. 6).

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR is based on the battery cell parameter values defined in Table 3.8.3-1. 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

Category A defines the normal parameter limit for each designated pilot cell in each battery. The chosen pilot cells are the weakest cells in the battery based on previous test results. These cells are monitored closely as an indication of battery performance.

The Category A limits specified for electrolyte level are based on manufacturer's recommendations and are consistent with the guidance in IEEE-450 (Ref. 7), with the extra 1/4" allowance above the high-water-level indication for

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

operating margin to account for temperatures and charge effects. In addition to this allowance, a footnote to Table 3.8.3-1 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 suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 7) 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 ≥ 2.13 V per cell. This value is based on the recommendations of IEEE-450 (Ref. 7), which state that prolonged operation of cells below 2.13 V can reduce the life expectancy of cells. Because resistivity decreases and the charging current increases as the temperature of electrolyte increases, in order to maintain a constant cell voltage, IEEE-450 states that if a warmer cell is below 2.13 V its voltage can be corrected by adding 0.003 V for each degree Fahrenheit (0.005 V/°C) that the cell temperature exceeds the average temperature of other cells. Nevertheless, considering that having dissimilar cell temperatures is an undesirable situation, it is not expected that this correction will have to be made. Instead, appropriate plant preventive actions should be established in order to eliminate the possible causes of the temperature differential.

The Category A limit specified for specific gravity for each pilot cell is ≥ [1.200] (0.015 below the manufacturer's 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. 7), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings shall be corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), add 1 point (0.001) to the reading; subtract 1 point for each 3°F below 77°F. The specific gravity of the electrolyte in a cell will increase with a loss of water due to electrolysis or evaporation. A Note in Table 3.8.3-1 requires the above-mentioned correction

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 [3 to 7] 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 of the designated pilot cell. This phenomenon is discussed in IEEE-450 (Ref. 7). A footnote to Table 3.8.3-1 allows the float charge current to be used as an alternate to specific gravity following a battery recharge.

Category B

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out because of a degraded condition or for any other reason.

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 ≥ [1.195] (0.020 below the manufacturer's fully charged, nominal specific gravity) with the average of all connected cells ≥ [1.205] (0.010 below the manufacturer's 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. A Note to Table 3.8.3-1 requires correction of specific gravity for electrolyte temperature and level. This level correction is not required when battery charging current is < [2] amps on float charge.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Category C

Category C defines the allowable values 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 allowable value, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C allowable values 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 allowable value for float voltage is based on IEEE-450 (Ref. 7), 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 allowable value of average specific gravity is based on manufacturer's recommendations (≥ [1.195], 0.020 below the manufacturer's 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 will not mask overall degradation of the battery. The Notes to Table 3.8.3-1 that apply to Category A specific gravity are also applicable to Category C specific gravity.

The SR to verify Category A cell parameters is consistent with IEEE-450 (Ref. 7), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of pilot cells. If pilot cells have one or more battery cell parameters not within Category A limits, the electrolyte level and float voltage of the pilot cells should be verified to meet Category C allowable values within 1 hour. This check will provide a quick indication of the status of the remainder of the battery cells. One hour provides sufficient time to inspect the electrolyte level and to

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

confirm the float voltage of the pilot cell. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C allowable values are met 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 required verification because specific-gravity measurements must be obtained for each connected cell. Taking into consideration 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.

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. Taking into consideration that while battery capacity is degraded, sufficient capacity exists to perform the intended function and allow time to fully restore the battery cell parameters to normal limits, this time is acceptable. When any battery parameter is outside the Category C allowable value 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.

SR 3.8.3.2

Verifying battery terminal voltage while on float charge for the [258/129] V 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's recommendations and IEEE-450 (Ref. 7).

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.3

This SR is based on the battery cell parameters defined in Table 3.8.3-1. The meaning of these different parameters is explained in SR 3.8.3.1 above. The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 7). 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. This inspection is also consistent with IEEE-450 (Ref. 7), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurred as a consequence of such discharge or overcharge. The steps to follow in case one or more battery cell parameters are not within limits are described above in SR 3.8.3.1.

SR 3.8.3.4

This Surveillance, verification that the average temperature of representative cells is ≥ [60°F], is consistent with a recommendation of IEEE-450 (Ref. 7), which states that the temperature of electrolytes in representative cells should be determined on a quarterly basis. IEEE-450 suggests taking the temperature of every sixth cell.

While higher-than-normal operating temperatures increase battery capacity, increase internal discharge, lower cell voltages for a given charge current, and raise charging current for a given charge voltage, they decrease battery life.

Lower-than-normal temperatures have the opposite effect, acting to inhibit or reduce battery capacity. Normal battery operating temperatures are [60°]F to [90°F], with a recommended operating temperature of [77]°F. This SR ensures that the operating temperatures remain within an acceptable operating range. These limits are based on manufacturer's recommendations.

SR 3.8.3.5

Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

inter-cell, inter-rack, inter-tier, and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The limits established for this SR 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 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. In addition, consistent with IEEE-450 (Ref. 7), SR 3.8.3.7 and SR 3.8.3.8 require yearly visual inspection, to detect corrosion, and yearly resistance measurements of connections.

SR 3.8.3.6

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.

This SR is consistent with IEEE-450 (Ref. 7), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.

SR 3.8.3.7 and SR 3.8.3.8

Visual inspection and resistance measurements of inter-cell, inter-rack, inter-tier, and terminal connections provides an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anti-corrosion 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 connection resistance limits are the same as those stated in SR 3.8.3.5 above.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequencies of 12 months are consistent with IEEE-450 (Ref. 7), which recommends detailed visual inspection of cell condition and inspection of cell-to-cell and terminal connection resistance on a yearly basis.

SR 3.8.3.9

This SR requires that each battery charger be capable of supplying [400] amps and [250/125] V for ≥ [8] hours. These requirements are based on the design capacity of the chargers (Ref. 4). According to Regulatory Guide 1.32 (Ref. 8), 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 ensures that these requirements can be satisfied. This Surveillance is required to be performed during {VS-BW,CE,W: MODES 5 and 6} {VS-GE: MODES 4 and 5} since it would require the DC electrical power subsystem to be inoperable during performance of the test.

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.

SR 3.8.3.10

A battery-service test is a special test of the battery's 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. Reference 4 provides load requirements for DC electrical power subsystems. [Optionally, the design duty-cycle requirements may be defined here].

The Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 8) and Regulatory Guide 1.129 (Ref. 9), which state that the battery-service test should be performed during refueling

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

operations or at some other outage, with intervals between tests not to exceed 18 months.

A Note to SR 3.8.3.10 allows the once-per-60-months performance of SR 3.8.3.11 in lieu of SR 3.8.3.10. This substitution is acceptable because SR 3.8.3.11 represents a more severe test of battery capacity than SR 3.8.3.10.

This Surveillance is required to be performed during {VS-BW,CE,W: MODES 5 and 6} {VS-GE: MODES 4 and 5} since it would require a DC electrical power subsystem to be inoperable during performance of the test.

SR 3.8.3.11

A battery-performance 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.

The acceptance criteria for this Surveillance is consistent with IEEE-450 (Ref. 7) 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 60 months, or every 12 months if the battery shows degradation or has reached 85% of its expected life. Degradation is indicated, according to IEEE-450 (Ref. 7), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is below the manufacturer's rating. An additional SR calls for a performance test on a newly installed battery within 24 months. All these Frequencies are consistent with the recommendations in IEEE-450 (Ref. 7).

This Surveillance is required to be performed during {VS-BW,CE,W: MODES 5 and 6} {VS-GE: MODES 4 and 5}, since

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

it would require the DC electrical power subsystem to be inoperable during performance of the test.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 17, "Electric Power System."

2. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," U.S. Nuclear Regulatory Commission, March 10, 1971.

3. IEEE-308 [1978], "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.

4. [Unit Name] FSAR, Section [ ], "[Title]."

5. IEEE-485 [1983], "Recommended Practices for Sizing Large Lead Storage Batteries for Generating Stations and Substations," Institute of Electrical and Electronic Engineers, June 1983.

6. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.

7. IEEE-450 [1987], "IEEE Recommended Practice for Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems," Institute of Electrical and Electronic Engineers.

8. Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," February 1977, U.S. Nuclear Regulatory Commission.

9. Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems," U.S. Nuclear Regulatory Commission, December 1974.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources—Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for Specification 3.8.3, "DC Sources—Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC and DC electrical power sources and associated distribution systems during shutdown and refueling, as specified in the LCO, ensures that (Ref. 1):

a. The facility 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 an inadvertent draindown of the vessel or a fuel-handling accident.

Although in many cases the FSAR may only address bounding analyses that are typically for power operation, for other Modes of operation (Ref. 2), among other requirements, are still required to be met. As these GDC are not MODE specific, and as it is a function of the Technical Specifications (TS) to ensure that the plant is operated within its design basis, with regard to DC sources, the requirements established in the TS must be consistent with the GDC related to electrical systems, as well as other GDC related to safety-related systems, since the DC sources comprise a typical support system.

In general, when the plant is shut down, the TS requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents assuming a single failure, because either:

a. Redundant and independent systems are required to be OPERABLE; or

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APPLICABLE SAFETY ANALYSES (continued)

b. Appropriate administrative measures are established and/or alternate backup systems that can provide functional redundant capability are required to be OPERABLE or put into operation in a period of time commensurate with the accident and the initial conditions considered.

This statement, in general, is reflected in the system LCOs for shutdown MODES of operation.

In addition to the postulated shutdown events directly addressed in the plant FSAR, it is necessary to consider evaluations of plant data that show that a large number of events can take place during shutdown. If not mitigated, some of these events can lead to core damage. Typically, the loss of decay-heat removal while there is substantial core decay heat poses a significant likelihood of a release due to a severe core damage accident.

To avoid the consequences of possible accidents during shutdown, different requirements are established according to the design of each plant. So, as far as residual heat removal (RHR) is concerned {VS-BW,CE,W: the OPERABILITY of the two RHR loops is required in MODES 5 and 6 when the reactor coolant loops are not filled (MODE 5) and when Reactor Coolant System (RCS) water level above the top of the reactor vessel flange is less than 23 feet (MODE 6). See Specifications 3.4.8, "RCS Loops—MODE 5, Loops Not Filled," and {VS-W: 3.9.7, "Residual Heat Removal and Coolant Circulation—Low Water Level."} {VS-CE: 3.9.5, "Shutdown Cooling and Coolant Circulation—Low Water Level."} {VS-BW: 3.9.5, "Decay Heat Removal and Coolant Circulation—Low Water Level."}} {VS-GE: the OPERABILITY of the two RHR shutdown cooling subsystems is always required in MODE 4 and in MODE 5 when RCS water level above the top of the reactor vessel flange is less than 23 feet. See Specifications {VS-BWR/4: 3.4.8,} {VS-BWR/6: 3.4.9,} "Residual Heat Removal Shutdown," and 3.9.8, "Residual Heat Removal—Low Water Level."} Therefore, in these conditions, [1 and 2] DC electrical power sources are required to be OPERABLE as support systems.

The DC Sources satisfy Criterion 3 of the NRC Interim Policy Statement.

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BASES (continued)

LCO

LCO 3.8.4.a requires OPERABILITY of the DC electrical power subsystem associated with the one [division] of the onsite Class 1E power distribution subsystem of LCO 3.8.8.a. The intent is that all required non-redundant loads, as well as one required load from each required redundant pair of loads, be powered from the same safety [division] and that all required AC and DC electrical power sources, as well as the power distribution subsystem itself, will be OPERABLE so that the AC and DC electrical power sources and power distribution subsystem will be capable of fully supporting the non-redundant loads.

When redundant counterpart loads (e.g., the second members of the pair) are required to be OPERABLE, LCO 3.8.4.b requires that they receive DC electrical power from the other [division] DC electrical power subsystem associated with the one [division] of the onsite Class 1E power distribution subsystem of LCO 3.8.8.b. Therefore, LCO 3.8.4.b requires this other [division] DC electrical power subsystem to be OPERABLE.

{VS-BWR/6: LCO 3.8.4.c requires OPERABILITY of the [division 3] DC electrical power subsystem associated with the onsite Class 1E power distribution subsystem of LCO 3.8.8.c when the High Pressure Core Spray (HPCS) System is required to be OPERABLE, or when other loads assigned to the HPCS system [division] are required to be OPERABLE, or both.}

See the Bases of Specification 3.8.3 for additional information on DC electrical power source OPERABILITY and DC electrical power source support and supported systems.

LCO 3.8.4 specifies the minimum number of DC sources required to be OPERABLE in MODES {VS-BW,CE,W: 5 and 6} {VS-GE: 4 and 5} and any time when handling irradiated fuel {VS-GE: [or moving loads over irradiated fuel in the primary or secondary containment]}. It ensures the availability of sufficient DC electrical power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel-handling accidents, inadvertent reactor vessel draindown).

(continued)

(continued)

BASES (continued)

LCO (continued)

As described in the previous section, "Applicable Safety Analyses," in the event of an accident during shutdown, the TS are designed to maintain the plant in such a condition that, even with a single failure, the plant will not be in immediate difficulty. In some cases, this is accomplished by requiring completely redundant and independent systems to be OPERABLE. In other cases, if justified based on a single plant design, administrative measures may be sufficient to relax the single-failure criterion. Also, an alternative backup system that provides the same functional capability may be substituted, provided the backup system is OPERABLE or can be made OPERABLE in sufficient time to mitigate the consequences of an accident during shutdown. When required to be OPERABLE, systems are reliable only if their support requirements are also met. The DC sources comprise a typical support system.

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES {VS-BW,CE,W: 5 and 6} {VS-GE: 4 and 5} and also any time when handling irradiated fuel {VS-GE: [or moving loads over irradiated fuel in the primary or secondary containment]} provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

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

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

DC electrical power requirements for {VS-BW,CE,W: MODES 1, 2, 3, and 4} {VS-GE: MODES 1, 2, and 3} are covered in Specification 3.8.3, "DC Sources—Operating."

(continued)

ACTIONS

A.1, A.2, A.3, A.4, A.5, and A.6

With one or more of the required DC electrical power subsystems inoperable, some equipment is not receiving the minimum support it needs. Therefore, it is required to suspend CORE ALTERATIONS, handling of irradiated fuel, {VS-GE: moving of loads over irradiated fuel,} any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions will preclude the occurrence of actions that could potentially initiate the 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's 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 the unit's safety systems may be without power.

Required Action A.6 verifies that the Required Actions for supported systems declared inoperable because of the inoperability of one or more DC electrical power subsystems have been initiated and within the same Completion Time as that specified for Required Action A.5.

Required Action A.6 ensures that identified Required Actions associated with supported systems affected by the inoperability of one or more DC electrical power subsystems have been initiated. This can be accomplished by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Action for Condition A of this LCO.]

[For this facility, the identified supported systems' Required Actions are as follows:]

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

SR 3.8.4.1 requires performance of all Surveillances required by SR 3.8.3.1 through SR 3.8.3.11. Therefore, see the corresponding Bases for Specification 3.8.3 for a discussion of each SR.

REFERENCES

1. [Unit Name] FSAR, Section [ ], "[Title]."
2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 Inverters—Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve in being powered from the DC battery source. There is [one] inverter per AC vital bus making a total of [four] inverters. The function of the inverter is to convert DC electrical power to AC electrical power, thus providing an uninterruptible power source for the instrumentation and controls for the Reactor Protection System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). The inverters are powered from the [120] V battery source.

[For this facility, specific background details on inverters, such as type, capacity, operating limits, and number and status of spares, are as follows:]

[ ]

APPLICABLE SAFETY ANALYSES

The initial conditions of design basis transient and accident analyses in [the FSAR, Chapter 6, "Engineered Safety Features," and Chapter 15, "Accident Analyses"], assume ESF 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 RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Technical Specifications 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

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 plant. This includes maintaining at least one [division] of the onsite or offsite AC electrical power sources, DC electrical power sources,

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(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	and associated distribution systems 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 satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

The power distribution subsystems listed in Table B 3.8.7-1 include the inverters. These inverters ensure the availability of AC electrical power for the instrumentation 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 LCO states that the required inverters shall be OPERABLE. The required inverters for [Division 1] are [Plant Specific: . . . fill in the inverter numbers for [Division 1]]. The required inverters for [Division 2] are [Plant Specific: . . . fill in the inverter numbers for [Division 2]].

{VS-BWR/6: [Division 3] inverters that support the High Pressure Core Spray (HPCS) System or both the HPCS System and other systems are required OPERABLE by LCO 3.8.5 if they are needed to ensure the OPERABILITY OF THE HPCS System and the other systems that they support.}

Upon the inoperability of one required inverter, Condition A is entered. Upon the inoperability of two or more required inverters, entry into LCO 3.0.3 is implicitly required.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is not defeated. If one required inverter is inoperable the possibility of a reactor spurious trip is increased. The [four] battery-powered inverters ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the [4.16 kV] safety buses are de-energized.

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BASES (continued)

LCO (continued)

OPERABILITY is met, as it applies to inverters, provided a correct DC voltage ([120] V) is applied, a correct AC voltage is at the output, and these voltages are within the design voltage and frequency tolerances. Furthermore, the inverters must be within the manufacturers' specifications for environmental factors such as temperature and humidity.

This LCO is modified by a Note allowing [two] inverters to be disconnected from their associated DC buses for ≤ 24 hours. This allowance is provided to perform an equalizing charge on one battery bank. If the inverters were not disconnected, the resulting voltage condition might damage the inverters energized [from their associated DC buses]. Disconnecting the inverters is allowed provided the associated AC vital buses are energized from their Class 1E constant voltage source transformer and the AC vital buses for other battery banks are energized from the associated inverters connected to their DC buses. 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. When utilizing the allowance, if one or more of the provisions is not met (e.g., 24-hour time period exceeded, etc.), LCO 3.0.3 must be entered immediately.

The intent of this Note is to allow only the [one] inverter[s] powered from [its/their] associated DC bus to be disconnected. [Thus, for plants with one battery bank per [division], two inverters may be disconnected. For plants with two battery banks per [division], only one inverter may be disconnected.]

[For this facility, an OPERABLE inverter constitutes the following:]

[For this facility, the following support systems are required OPERABLE to ensure inverter OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not declare inverters inoperable and their justification are as follows:]

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BASES (continued)

LCO (continued)

[For this facility, the supported systems affected by the inoperability of an inverter and the justification for whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

The inverters are required to be OPERABLE in {VS-BW,CE,W: MODES 1, 2, 3, and 4} {VS-GE: MODES 1, 2, and 3} 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 {VS-BW,CE,W: MODES 5 and 6} {VS-GE: MODES 4 and 5} are covered in the Bases for Specification 3.8.6.

ACTIONS

A.1, A.2, A.3, and A.4

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. Required Action A.1 allows up to 2 hours to perform this task {VS-BWR/6: OR [2 hours] if a [Division 3] inverter is the inoperable inverter}.

{VS-BWR/6: [The [2-hour] Completion Time for an inoperable [division 3] inverter is plant specific. Items to be considered in specifying this Completion Time for a given facility include:

a. The safety function of [Division 3]. If [Division 3] supports other ESF functions in addition to the HPCS function, then the Completion Time for an inoperable [Division 3] inverter shall be [2 hours]; and

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BASES (continued)

ACTIONS (continued)

b. The safety function of the [Division 3] inverter(s):

1. What is its role in mitigating a DBA?

2. What systems does it support?

3. What is its role in mitigating a station blackout?])

The 2-hour Completion Time is consistent with the 2-hour Completion Time for an inoperable DC bus, and an inoperable AC vital bus (see Specification 3.8.7, "Distribution Systems—Operating"). Required Actions A.2 and A.3 allow 24 hours to fix the inoperable inverter and return it to service (VS-BWR/6: OR [24 hours] if a [division 3] inverter is the inoperable inverter. [The [24-hour] Completion Time is plant specific, and the items listed above should be considered in specifying this time for a given facility]). 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 plant 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). Thus, the probability of a spurious reactor trip is increased. Similarly, the uninterruptible, battery-backed, inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices, because the constant voltage transformer source is more susceptible to voltage drift/degraded voltage than is the inverter source to the AC vital buses.

Required Action A.4 verifies that the Required Actions for those supported systems declared inoperable because of the inoperability of one inverter have been initiated and within the same Completion Time as that of Required Action A.1.

Required Action A.4 ensures that those identified Required Actions associated with supported systems affected by the inoperability of the inverter have been initiated. This can be accomplished by entering the supported systems' LCOs.

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BASES (continued)

ACTIONS (continued)

[Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Action for Condition A of this LCO.]

[For this facility, the identified support systems' Required Actions are as follows:]

B.1

With one required inverter inoperable AND one or more support or supported features, or both, inoperable associated with the other OPERABLE inverters, or with opposite OPERABLE AC and DC electrical power distribution subsystems, or with opposite OPERABLE DC electrical power subsystems, or all three, there is a loss of functional capability and LCO 3.0.3 must be immediately entered. However, if the LCOs of the support or supported feature, or both, take into consideration the loss of function situation, then LCO 3.0.3 may not need to be entered.

C.1 and C.2

The plant must be placed in a MODE in which the LCO does not apply if the inoperable devices or components cannot be restored to OPERABLE status within the associated Completion Time. This is done by placing the plant in at least MODE 3 within {VS-BW,CE,W: 6 hours} {VS-GE: 12 hours} and in {VS-BW,CE,W: MODE 5} {VS-GE: MODE 4} within 36 hours. The allowed Completion Times are reasonable, based on operating experience related to the amount of time required to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	redundant capability of the inverters and other indications available in the control room that will alert the operator to inverter malfunctions.
REFERENCES	None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Inverters—Shutdown

BASES

BACKGROUND	A description of the inverters is provided in the Bases for Specification 3.8.5, "Inverters—Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources, DC sources, and inverter sources to each AC vital bus during shutdown and refueling, as specified in the LCO, ensures that (Ref. 1):

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel-handling accident.

In particular, instrumentation and control capability is powered from the AC vital buses, which are themselves powered by the inverters.

Although in many cases the FSAR may only address bounding analyses that are typically for power operation, for other modes of operation, the GDC (Ref. 2), among other requirements are still required to be met. As these GDC are not MODE specific, and as it is a function of the Technical Specifications (TS) to assure that the plant is operated within its design basis, with regard to AC sources, DC sources, and inverters, the requirements established in the TS must be consistent with the GDC related to electrical systems, as well as with other GDC related to safety-related systems, since the AC sources, DC sources, and inverters are typical support systems.

In general, when the plant is shut down, the TS requirements ensure that the plant has the capability to mitigate the

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

consequences of postulated accidents assuming a single failure, because either:

a. Redundant and independent systems are required to be OPERABLE, or

b. Appropriate administrative measures are established and/or alternate backup systems that can provide functional redundant capability are required to be OPERABLE or put into operation in a period of time commensurate with the accident and the initial conditions considered.

This statement, in general, is reflected in the system LCOs for shutdown MODES of operation.

In addition to the postulated shutdown events directly addressed in the plant FSAR, it is necessary to consider evaluations of plant data that show a large number of events can take place during shutdown. If not mitigated, some of these events can lead to core damage. Typically, the loss of decay-heat removal while there is substantial core decay heat poses a significant likelihood of a release due to a severe core damage accident.

To avoid the consequences of possible accidents during shutdown, different requirements are established according to the design of each plant. So, as far as residual heat removal (RHR) is concerned {VS-BW,CE,W: the OPERABILITY of the two RHR loops is required in MODES 5 and 6 when the reactor coolant loops are not filled (MODE 5) and when the Reactor Coolant System (RCS) water level above the top of the reactor vessel flange is less than 23 feet (MODE 6). See Specifications 3.4.8, "RCS Loops—MODE 5, Loops Not Filled," and {VS-W: 3.9.7, "Residual Heat Removal and Coolant Circulation—Low Water Level."} {VS-CE: 3.9.5, "Shutdown Cooling and Coolant Circulation—Low Water Level."} {VS-BW: 3.9.5, "Decay Heat Removal and Coolant Circulation—Low Water Level."} {VS-GE: the OPERABILITY of the two RHR shutdown cooling subsystems is always required in MODE 4, and in MODE 5 when RCS water level above the top of the reactor vessel flange is less than 23 feet. See Specifications {VS-BWR/4: 3.4.8,} {VS-BWR/6: 3.4.9,} "Residual Heat Removal—Shutdown," and 3.9.8, "Residual

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	Heat Removal—Low Water Level.") Therefore, in these conditions, [Division 1 and 2] inverter sources to the AC vital buses are required to be OPERABLE as support systems. The inverters satisfy Criterion 3 of the NRC Interim Policy Statement.
LCO	LCO 3.8.6.a requires OPERABILITY of the inverters associated with the one [division] of the onsite Class 1E power distribution subsystem of LCO 3.8.8.a. The intent is that all required non-redundant loads, as well as one required load from each required redundant pair of loads, be powered from the same safety [division] and that all required AC, DC, and inverter sources, as well as the distribution subsystem itself, will be OPERABLE so that the AC, DC, and inverter sources and the distribution subsystem will be capable of fully supporting the non-redundant loads. When redundant counterpart loads (e.g., the second members of the pair) are required to be OPERABLE, LCO 3.8.6.b requires that they receive inverter support from the other [division] inverters associated with the one [division] of the onsite Class 1E power distribution subsystem of LCO 3.8.8.b. Therefore, LCO 3.8.6.b requires this other [division] inverters to be OPERABLE. {VS-BWR/6: LCO 3.8.6.c requires OPERABILITY of the [Division 3] inverters associated with the onsite Class 1E power distribution subsystem of LCO 3.8.8.c when the High Pressure Core Spray (HPCS) System is required to be OPERABLE, or when other loads assigned to the HPCS System [division] are required to be OPERABLE, or both.} See the Bases for Specification 3.8.5 for additional information on inverter OPERABILITY, and inverter support and supported systems. LCO 3.8.6 specifies the minimum number of inverters required to be OPERABLE in MODES {VS-BW,CE,W: 5 and 6} {VS-GE: 4 and 5} and any time when handling irradiated fuel {VS-GE: [or moving loads over irradiated fuel in the primary or secondary containment]}. It ensures the availability of (continued)

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BASES (continued)

LCO (continued)

sufficient inverter power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel-handling accidents, inadvertent reactor vessel draindown).

As described in the previous section, "Applicable Safety Analyses," in the event of an accident during shutdown, the TS are designed to maintain the plant in a condition so that even with a single failure, the plant will not be in immediate difficulty. In some cases, this is accomplished by requiring completely redundant and independent systems to be OPERABLE. In other cases, if justified based on a single plant design, administrative measures may be sufficient to relax the single-failure criterion. Also, an alternative backup system that provides the same functional capability may be substituted provided the backup system is OPERABLE, or can be made OPERABLE in sufficient time to mitigate the consequences of an accident during shutdown. When required to be OPERABLE, systems are reliable only if their support requirements are also met. The inverters comprise a typical support system.

APPLICABILITY

The inverters required to be OPERABLE in MODES {VS-BW,CE,W: 5 and 6} {VS-GE: 4 and 5} and also any time when handling irradiated fuel {VS-GE: [or moving loads over irradiated fuel in the primary or secondary containment]} provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

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

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

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BASES (continued)

APPLICABILITY (continued)

Inverter requirements for {VS-BW,CE,W: MODES 1, 2, 3, and 4} {VS-GE: MODES 1, 2, and 3} are covered in Specification 3.8.5, "Inverters—Operating."

ACTIONS

A.1, A.2, A.3, A.4, A.5, and A.6

With one or more of the required inverters inoperable, some equipment is not receiving the minimum support it needs. Therefore, it is required to suspend CORE ALTERATIONS, handling of irradiated fuel {VS-GE: moving of loads over irradiated fuel,} any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions will preclude the occurrence of actions that could potentially initiate the 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's 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's safety systems may be without power or powered from a constant voltage source transformer.

Required Action A.6 verifies that the Required Actions for those supported systems declared inoperable because of the inoperability of one or more inverters have been initiated and within the same Completion Time as that specified for Required Action A.5.

Required Action A.6 ensures that identified Required Actions associated with supported systems affected by the inoperability of one or more inverters have been initiated.

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BASES (continued)

ACTIONS (continued)	This can be accomplished by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition A of this LCO.] [For this facility, the identified supported systems Required Actions are as follows:]
SURVEILLANCE REQUIREMENTS	SR 3.8.6.1 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 Reactor Protection System and Engineered Safety Feature Actuation System 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 will alert the operator to inverter malfunctions.
REFERENCES	1. [Unit name] FSAR, Section [ ], "[Title]." 2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Distribution Systems—Operating

BASES

BACKGROUND

{VS-BW,CE,W,BWR/4: The onsite Class 1E AC and DC electrical power distribution system is divided by [division] into [two] redundant and independent AC and DC electrical power distribution subsystems. Each [divisional] AC and DC electrical power distribution subsystem is comprised of [PLANT SPECIFIC: List the major AC, AC vital, and DC bus names used in Table B 3.8.7-1. For example: 4.16 kVac ENGINEERED SAFETY FEATURE (ESF) buses, 480 Vac load centers, buses, motor control centers, and 120 Vac power distribution panels; 120 Vac vital buses; and 250/125 Vdc buses]. [Two] [divisions] (or subsystems) are required for safety function redundancy; [any one] [division] (or subsystem) provides safety function, but without worst-case single-failure protection.}

{VS-BWR/6: The onsite Class 1E AC and DC electrical power distribution system is divided by [division] into [three] independent AC and DC electrical power distribution subsystems. Each [divisional AC and DC electrical power distribution subsystem is comprised of [PLANT SPECIFIC: List the major AC, AC vital, and DC bus names used in Table B 3.8.7-1. For example: 4.16 kVac ESF buses, 480 Vac load centers, buses, motor control centers, and 120 Vac power distribution panels; 120 Vac vital buses; and 250/125 Vdc buses]. All three [divisions] (or subsystems) are required for safety function redundancy; any two [divisions] (or subsystems) provide safety function, but without worst-case single-failure protection.}

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 source. Each [4.16 kV ESF bus] is normally connected to a preferred source. During a loss of one offsite power source to the [4.16 kV ESF buses], a [4.16 kV] transfer scheme is accomplished by utilizing a time-delayed bus undervoltage relay. If all offsite sources are unavailable, the onsite emergency power system will supply power to the [4.16 kV ESF buses]. Control power for the [4.16 kV breakers] is supplied from the [Class 1E batteries]. Additional description of this system may be

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BASES (continued)

BACKGROUND (continued)

found in the Bases for Specification 3.8.1, "AC Sources—Operating," and the Bases for Specification 3.8.3, "DC Sources—Operating."

The secondary plant distribution is at [480] V. The [480] V distribution system includes [PLANT SPECIFIC: List items such as emergency buses, load centers, and transformers; the identifying numbers of these items should also be included]. The [480] V load centers from each subsystem are located [in separate rooms in the control building]. Control power for the [480] V breakers is supplied from the [Class 1E batteries], as described in the Bases for Specification 3.8.3, "DC Sources—Operating."

The Class 1E [480] Vac motor control centers and power distribution panels are powered from [PLANT SPECIFIC: Provide bus and/or load center information and nomenclature].

The Class 1E [120] V power distribution panels are powered from [PLANT SPECIFIC: Provide distribution panel information and nomenclature]. All [120] V distribution panels that provide control or instrumentation necessary for operation of safety systems are required to be included in this specification.

The [120] Vac vital buses [2YV1, 2YV2, 2YV3, and 2YV4] are arranged in four load groups and are normally powered from [PLANT SPECIFIC: Provide power path and nomenclature between the inverters and the buses]. The alternate power supply for the vital buses is a [Class 1E constant voltage source transformer] powered from the same [division] as the associated inverter, and its use is governed by LCO 3.8.5, "Inverters—Operating." Each constant voltage source transformer is powered from [PLANT SPECIFIC: Provide power path and nomenclature].

There are {VS-BW,CE,W,BWR/4: [two]} {VS-BWR/6: [three]} independent [125/250] Vdc electrical power distribution subsystems. [PLANT SPECIFIC: Provide power path and nomenclature for the DC power distribution system.]

The list of all required distribution buses is located in Table B 3.8.7-1.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The initial conditions of design basis transient and accident analyses in [FSAR Chapter 6, "Engineering Safety Features," and Chapter 15, "Accident Analyses,"] assume ESF systems are OPERABLE. The AC and DC electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS) and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Technical Specifications 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

The OPERABILITY of the AC and DC electrical power distribution system is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one [division] of the onsite or offsite AC electrical power sources, DC electrical power sources, and associated 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 AC and DC electrical power distribution system satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The required AC and DC [divisional] power distribution subsystems listed in Table B 3.8.7-1 ensure the availability of AC and DC electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The {VS-BW,CE,W,BWR/4: [Division 1 and 2]} {VS-BWR/6: [Division 1, 2, and 3]} AC and DC electrical power distribution subsystems are required to be OPERABLE.

{VS-BW,CE,W,BWR/4: Maintaining the [Division 1 and 2] AC and DC electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Either [division] of the AC and DC

(continued)

(continued)

BASES (continued)

LCO
(continued)

power distribution system is capable of providing the necessary electrical power to the associated ESF components. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.}

{VS-BWR/6: Maintaining th [Division 1, 2, and 3] AC and DC electrical power distrib on subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. [Any two of the three] [divisions] of the distribution system are capable of providing the necessary electrical power to the associated ESF components. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.}

OPERABILITY is met, as it applies to AC and DC electrical power distribution subsystems, provided the associated buses, transformers, load centers, motor control centers, and electrical circuits are fully energized to their proper voltages and frequencies. The components of each AC and DC electrical power distribution subsystem must be kept within the manufacturers' specifications for environmental factors such as temperature and humidity.

In addition, breakers must be open between redundant buses to prevent two power sources from being paralleled. The open breakers also preclude unlimited continued operation where a single failure (loss of one source) could cause a loss of two redundant buses. Thus, if two sources are paralleled through redundant distribution buses that are cross-tied, the distribution buses must be considered inoperable. If two redundant buses are powered from the same source, however, only the bus that is not being powered from its normal source shall be considered inoperable.

[For this facility, as a minimum, the following support systems associated with the AC and DC electrical power distribution subsystems governed by LCO 3.8.7 to ensure their OPERABILITY are as follows:]

[For this facility, the supported systems affected by the inoperability of the support systems governed by LCO 3.8.7, and the justification of whether or not each supported system is declared inoperable, are as follows:]

(continued)

BASES (continued)

APPLICABILITY

The AC and DC electrical power distribution subsystems are required to be OPERABLE in {VS-BW,CE,W: MODES 1, 2, 3, and 4} {VS-GE: MODES 1, 2, and 3} 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.

AC and DC electrical power distribution subsystem requirements for {VS-BW,CE,W: MODES 5 and 6} {VS-GE: MODES 4 and 5} are covered in the Bases for Specification 3.8.8.

A Note has been added to provide clarification that for this LCO, all required AC and DC electrical power distribution subsystems shall be treated as an entity with a single Completion Time.

ACTIONS

A.1

With one or more required AC buses, load centers, motor control centers, or distribution panels, except AC vital buses, in one division inoperable the remaining AC electrical power distribution {VS-BW,CE,W,BWR/4: subsystem is} {VS-BWR/6: 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 power distribution {VS-BW,CE,W,BWR/4: subsystem} {VS-BWR/6: 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 a determined amount of time ([ ] hours), not to exceed 8 hours if more than two systems are made inoperable because of the distribution system inoperability.

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BASES (continued)

ACTIONS (continued)

[ ] hours will be a specific number for each specific bus in each specific plant. For a specific bus, [ ] hours is defined as the most limiting Completion Time of all the supported systems that are made inoperable by the inoperability of the bus. Thus, a prior determination must be made to obtain the most limiting Completion Time of all the systems supported by each bus. [ ] does not exceed 8 hours, however, if three or more systems are made inoperable by the bus inoperability.

Note that the equipment referred to is all in one [division] power distribution subsystem.

When equipment governed by LCO 3.8.7 is inoperable in {VS-BW,CE,W,BWR/4: both [divisions]} {VS-BWR/6: two or more [divisions]} and results in loss of functional capability, then LCO 3.0.3 must be immediately entered.

B.1

With one 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. For an AC vital bus to be considered OPERABLE, it must be powered from its DC-to-AC inverter. An alternate Class 1E constant voltage source may be used if approved for this purpose as stated in the licensing basis of the plant. Requirements imposed on the alternate source are governed by LCO 3.8.5, "Inverters—Operating." 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.

{VS-BWR/6: However, if the inoperable AC vital bus is associated with [Division 3], then continued operation for up to a [2-hour] Completion Time is plant specific and is meant to be the most limiting Completion Time for all systems that a [Division 3] AC vital bus supports;

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BASES (continued)

ACTIONS (continued)

furthermore, the [2-hour] Completion Time is not to exceed 8 hours if more than two systems are made inoperable because of the [Division 3] AC vital bus inoperability. The [2-hour] Completion Time for [Division 3] takes into account the importance to safety of restoring the [Division 3] 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.}

When more than one AC vital bus is inoperable, there is a loss of functional capability. Therefore, LCO 3.0.3 must be immediately entered.

C.1

With one or more required DC buses in one [division] inoperable the remaining DC electrical power distribution {VS-BW,CE,W,BWR/4: subsystem is} {VS-BWR/6: 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 {VS-BW,CE,W,BWR/4: subsystem} {VS-BWR/6: subsystems} 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. The 2-hour Completion Time for DC buses is consistent with Regulatory Guide 1.93, "Availability of Electric Power Sources" (Ref. [1]).

{VS-BWR/6: However, if the inoperable DC bus is associated with [Division 3], then continued operation for up to a [2-hour] Completion Time is plant specific and is meant to be the most limiting Completion Time for all systems that a [Division 3] DC bus supports; furthermore, the [2-hour] Completion Time is not to exceed 8 hours if more than two systems are made inoperable because of the [Division 3] DC bus inoperability. The [2-hour] Completion Time for [Division 3] takes into account the importance to safety of restoring the [Division 3] DC bus to OPERABLE status, the redundant capability afforded by the other OPERABLE DC buses, and the low probability of a DBA occurring during this period.}

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BASES (continued)

ACTIONS (continued)

When one or more DC buses are inoperable in more than one AC and DC electrical power distribution subsystem, there is a loss of functional capability. Therefore, LCO 3.0.3 must be immediately entered.

D.1

With one or more features specified under Condition A, B, or C inoperable in the one [division] of AC and DC electrical power distribution subsystem AND one or more required support or supported features, or both, inoperable associated with the other OPERABLE AC and DC electrical power distribution subsystem(s), or with opposite OPERABLE DC electrical power subsystem(s), or both, there is a loss of functional capability and LCO 3.0.3 must be immediately entered. However, if the LCOs of the support or supported feature, or both, takes into consideration the loss of function situation, LCO 3.0.3 may not need to be entered.

E.1

With one or more features specified under Condition A, B, or C inoperable in one [division] of AC and DC electrical power distribution subsystem, verify that the Required Actions for those supported systems declared inoperable by the support features governed by LCO 3.8.7 have been initiated and within a Completion Time of [ ] hours.

The [ ]-hour Completion Time is defined as the most limiting of all the Required Actions for all the supported systems that need to be declared inoperable upon the failure of one or more features specified under Condition E.

Required Action E.1 ensures that those identified Required Actions associated with supported systems affected by the inoperability of the supported features governed by this LCO have been initiated. This can be accomplished by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition E of this LCO.]

[For this facility, the identified supported systems Required Actions are as follows:]

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BASES (continued)

ACTIONS (continued)

F.1 and F.2

The plant must be placed in a MODE in which the LCO does not apply if the inoperable devices or components cannot be restored to OPERABLE status within the associated Completion Time. This is done by placing the plant in at least MODE 3 within (VS-BW,CE,W: 6 hours) (VS-GE: 12 hours) and in (VS-BW,CE,W: MODE 5) (VS-GE: MODE 4) within 36 hours. The allowed Completion Times are reasonable, based on operating experience related to the amount of time required to reach the required MODES from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the AC and DC electrical power distribution systems are functioning properly, with all the required circuit breakers closed and the buses energized from normal power. 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 redundant capability of the AC and DC electrical power distribution subsystems, and other indications available in the control room that will alert the operator to subsystem malfunctions.

SR 3.8.7.2

This Surveillance verifies that the frequency on the AC vital buses is within limits. [For this facility, the purpose of this Surveillance is as follows:.

The 7-day Frequency takes into account the redundant capability of the AC and DC electrical power distribution subsystems and other indications available in the control room that will alert the operator to subsystem malfunctions.

REFERENCES

1. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.

{This version of Table B 3.8.7-1 is VS-BW,CE,W,BWR/4}

Table B 3.8.7-1 (page 1 of 1)

AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	[Division 1]*	[Division 2]*
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] from battery [NK11] and charger [NK21]	Bus [NK02] from battery [NK12] and charger [NK22]
		Bus [NK03] from battery [NK13] and charger [NK23]	Bus [NK04] from battery [NK14] and charger [NK24]
		Distribution Panels [NK41, NK43, NK51]	Distribution Panels [NK42, NK44, NK52]
AC vital buses	[120 V]	Bus [NN01] from inverter [NN11] connected to bus [NK01]	Bus [NN02] from inverter [NN12] connected to bus [NK02]
		Bus [NN03] from inverter [NN13] connected to bus [NK03]	Bus [NN04] from inverter [NN14] connected to bus [NK04]

* Each [division] of the AC and DC electrical power distribution system is a subsystem.

(This version of Table B 3.8.7-1 is VS-BW,CE,W,BWR/6)

Table B 3.8.7-1 (page 1 of 1)

AC and DC Electrical Power Distribution System

TYPE	VOLTAGE	[Division 1]*	[Division 2]*	[Division 3]*
AC safety buses	[4160 V]	[ESF Bus] [NB01]	[ESF Bus] [NB02]	[ESF Bus] [NB03]
	[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]	Motor Control Centers [NG05A, NG05C]
	[120 V]	Distribution Panels [NP01, NP03]	Distribution Panels [NP02, NP04]	Distribution Panels [NP05, NP06]
DC buses	[125 V]	Bus [NK01] from battery [NN11] and charger [NK21]	Bus [NK02] from battery [NK12] and charger [NK22]	Bus [NK05] from battery [NK15] and charger [NK25]
		Bus [NK03] from battery [NN13] and charger [NK23]	Bus [NK04] from battery [NK14] and charger [NK24]	
		Distribution Panels [NK41, NK43, NK51]	Distribution Panels [NK42, NK44, NK52]	Distribution Panel [NK45]
AC vital buses	[120 V]	Bus [NN01] from inverter [NN11] connected to bus [NK01]	Bus [NN02] from inverter [NN12] connected to bus [NK02]	Bus [NN05] from inverter [NN15] connected to bus [NK05]
		Bus [NN03] from inverter [NN13] connected to bus [NK03]	Bus [NN04] from inverter [NN14] connected to bus [NK04]	

* Each [division] of the AC and DC power distribution system is a subsystem.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution System—Shutdown

BASES

BACKGROUND

A description of the AC and DC electrical power distribution system is provided in the Bases for Specification 3.8.7, "Distribution System—Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC and DC electrical power sources and associated power distribution subsystems during shutdown and refueling, as specified in the LCO, ensures that (Ref. 1):

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel-handling accident.

Although in many cases the FSAR may only address bounding analyses that are typically for power operation, for other modes of operation, the GDC (Ref. 2), among other requirements, are still required to be met. As these GDC are not MODE specific, and as it is a function of the Technical Specifications (TS) to ensure that the plant is operated within its design basis, with regard to distribution systems, the requirements established in the TS must be consistent with the GDC related to electrical systems, as well as with other GDC related to safety-related systems, since the AC and DC electrical power distribution subsystems comprise a typical support system.

In general, when the plant is shut down, the TS requirements ensure that the plant has the capability to mitigate the

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

consequences of postulated accidents assuming a single failure, because either:

a. Redundant and independent systems are required to be OPERABLE; or

b. Appropriate administrative measures are established and/or alternate backup systems that can provide functional redundant capability are required to be OPERABLE or put into operation in a period of time commensurate with the accident and the initial conditions considered.

This statement, in general, is reflected in the system LCOs for shutdown MODES of operation.

In addition to the postulated shutdown events directly addressed in the plant FSAR, it is necessary to consider evaluations of plant data that show a large number of events can take place during shutdown. If not mitigated, some of these events can lead to core damage. Typically, the loss of decay-heat removal while there is substantial core decay heat poses a significant likelihood of a release due to a severe core damage accident.

To avoid the consequences of possible accidents during shutdown, different requirements are established according to the design of each plant. So, as far as residual heat removal (RHR) is concerned {VS-BW,CE,W,: the OPERABILITY of the two RHR loops is required in MODES 5 and 6 when the reactor coolant loops are not filled (MODE 5) and when the Reactor Coolant System (RCS) water level above the top of the reactor vessel flange is less than 23 feet (MODE 6). See Specifications 3.4.8, "RCS Loops—MODE 5, Loops Not Filled," and {VS-W: 3.9.7, "Residual Heat Removal and Coolant Circulation—Low Water Level."} {VS-CE: 3.9.5, "Shutdown Cooling and Coolant Circulation—Low Water Level."} {VS-BW: 3.9.5, "Decay Heat Remo l and Coolant Circulation—Low Water Level."}} {VS-GE: e OPERABILITY of the two RHR shutdown cooling subsystems is ways required in MODE 4, and in MODE 5 when RCS water le above the top of the reactor vessel flange is less than feet. See Specifications [VS-BWR/4: 3.4.8,} {VS-BWR/6: 3.4.9,} "Residual heat Removal—Shutdown," and 3.9.8, "Residual

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(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Heat Removal—Low Water Level.") Therefore, in these conditions, [portions of] [Division 1 and 2] AC and DC electrical power distribution subsystems are required to be OPERABLE as support systems.

The AC and DC electrical power distribution system satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

LCO 3.8.8.a requires OPERABILITY of one [division] AC and DC electrical power distribution subsystem. The intent is that all required non-redundant loads, as well as one required load from each required redundant pair of loads, be powered from this safety [division] and that all required AC and DC sources, as well as the distribution subsystem itself, will be OPERABLE so that the AC and DC sources and distribution subsystem will be capable of fully supporting the required loads.

When redundant counterpart loads (e.g., the second members of the pair) are required to be OPERABLE, LCO 3.8.8.b requires that they receive power from the [necessary portions of the] other [division] AC and DC electrical power distribution subsystem. Therefore, LCO 3.8.8.b requires [the necessary portions of] this other [division] DC electrical power subsystem to be OPERABLE.

{VS-BWR/6: LCO 3.8.8.c requires OPERABILITY of the [division 3] AC and DC electrical power distribution subsystem when the High Pressure Core Spray (HPCS) System is required to be OPERABLE, or when other loads assigned to the HPCS System [division] are required to be OPERABLE, or both.}

See the Bases for Specification 3.8.7 for additional information on AC and DC electrical power distribution subsystem OPERABILITY and AC and DC electrical power distribution support and supported systems.

LCO 3.8.8 specifies the minimum number of AC and DC electrical power distribution subsystems required to be OPERABLE in MODES {VS-BW,CE,W: 5 and 6} {VS-GE: 4 and 5} and any time when handling irradiated fuel {VS-GE: [or

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BASES (continued)

LCO (continued)

moving loads over irradiated fuel in the primary or secondary containment]}. It ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel-handling accidents, inadvertent reactor vessel draindown).

As described in the previous section, "Applicable Safety Analyses," in the event of an accident during shutdown, the TS are designed to maintain the plant in a condition such that, even with a single failure, the plant will not be in immediate difficulty. In some cases, this is accomplished by requiring completely redundant and independent systems to be OPERABLE. In other cases, if justified based on a single plant design, administrative measures may be sufficient to relax the single-failure criterion. Also, an alternative backup system that provides the same functional capability may be substituted provided the backup system is OPERABLE or can be made OPERABLE in sufficient time to mitigate the consequences of an accident during shutdown. When required to be OPERABLE, systems are reliable only if their support requirements are also met. The AC and DC electrical power distribution subsystems comprise a typical support system.

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES {VS-BW,CE,W: 5 and 6} {VS-GE: 4 and 5} and also any time when handling irradiated fuel {VS-GE: [or moving loads over irradiated fuel in the primary or secondary containment]} provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

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

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE; and

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BASES (continued)

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

AC and DC electrical power distribution subsystem requirements for {VS-BW,CE,W: MODES 1, 2, 3, and 4} {VS-GE: MODES 1, 2, and 3} are covered in Specification 3.8.7, "Distribution System—Operating."

ACTIONS

A.1, A.2, A.3, A.4, A.5, and A.6

With one or more of the required AC and DC electrical power distribution subsystems inoperable, some equipment is not receiving the minimum support it needs. Therefore, it is required to suspend CORE ALTERATIONS, handling of irradiated fuel, {VS-GE: moving of loads over irradiated fuel,} any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions will preclude the occurrence of actions that could potentially initiate the 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's safety systems.

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

Required Action A.6 verifies that the Required Actions for those supported systems declared inoperable because of the inoperability of one or more AC and DC electrical power distribution subsystems have been initiated and within the same Completion Time as that specified for Required Action A.5.

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BASES (continued)

ACTIONS (continued)

Required Action A.6 ensures that those identified Required Actions associated with supported systems affected by the inoperability of one or more AC and DC electrical power distribution subsystems have been initiated. This can be accomplished by entering the supported systems' LCOs. [Alternatively, the appropriate Required Actions for the supported systems may be listed in the Required Actions for Condition A of this LCO.]

[For this facility, the identified supported systems' Required Actions are as follows:]

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the AC and DC electrical power distribution system is functioning properly, with all required circuit breakers closed and the buses energized from normal power. 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 redundant capability of the AC and DC electrical power distribution subsystems, as well as other indications available in the control room that will alert the operator to subsystem malfunctions.

SR 3.8.8.2

This Surveillance verifies that the frequency on the AC vital buses is within limits. [For this facility, the purpose of this Surveillance is as follows:]

The 7-day Frequency takes into account the redundant capability of the AC and DC electrical power distribution subsystems, as well as other indications available in the control room that will alert the operator to subsystem malfunctions.

REFERENCES

1. [Unit name] FSAR, Section [ ], "[Title]."

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

B.3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling cavity, and the refueling canal 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 CORE OPERATING LIMITS REPORT (COLR). Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $K_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is 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 cavity and the refueling canal are then flooded with borated water from the refueling water storage tank (RWST) through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

During refueling, the water volumes in the RCS, the refueling cavity, and the refueling canal are contiguous.

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(continued)

BASES (continued)

BACKGROUND (continued)	However, the soluble boron concentration is not necessarily the same in each volume. If additions of boron are required during refueling, the CVCS makes it available through the RCS. The pumping action of the RHR System in the RCS, and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5 and LCO 3.9.6) to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling cavity, and the refueling canal above the COLR limit.
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). It includes an uncertainty allowance of [ ] ppm. The required boron concentration and the plant refueling procedures that verify the correct fuel-loading plan (including full-core mapping) ensure that the K_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling. The limiting boron dilution accident occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.2, "SHUTDOWN MARGIN - $T_{avg} \leq 200°F$." The RCS boron concentration in MODE 6 satisfies Criterion 2 of the NRC Interim Policy Statement.
LCO	The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling cavity, and the refueling canal while in MODE 6. The boron concentration limit specified in the COLR ensures that a core $K_{eff} \leq 0.95$

(continued)

(continued)

BASES (continued)

LCO (continued)	is maintained during fuel-handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.
APPLICABILITY	This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $K_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1 and LCO 3.1.2, "SHUTDOWN MARGIN," ensure that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

ACTIONS

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

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

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude completion of actions to establish a safe condition.

<u>A.3</u>

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated within 15 minutes. The 15-minute Completion Time is allowed for an operator to correctly align and start the required systems.

In determining the required combination of boration flow rate and concentration, no unique design basis event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the boration is performed at ≥ [ ] gpm of a solution containing [ ] ppm boron or its equivalent.

(continued)

(continued)

BASES (continued)

ACTIONS (continued)

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

In the event that the required boron concentration channels are found inoperable, the boron concentration is considered to be not within limits and Required Actions A.1, A.2, and A.3 apply.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR verifies that the coolant boron concentrations in the RCS, the refueling cavity, and the refueling canal are within the COLR limit. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

[This facility maintains the following controls to ensure that the likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote:]. A minimum Frequency of once every 72 hours is, therefore, a reasonable interval to verify the boron concentrations of representative samples. The surveillance interval is based on operating experience, which has shown 72 hours to be an adequate interval.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, Section VI, GDC 26, "Reactivity Control System Redundancy and Capability."

2. [Unit Name] FSAR, Section [ ], "[Accident Analysis]."

B 3.9 REFUELING OPERATIONS

B 3.9.2 Unborated Water Source Isolation Valves

BASES

BACKGROUND	During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water which are connected to the Reactor Coolant System (RCS) must be closed to prevent inadvertent dilution of the reactor coolant. The isolation valves must be secured in the closed position. The Chemical and Volume Control System (CVCS) is capable of supplying borated and unborated water to the reactor coolant system through various flow paths. Since positive reactivity additions made by reducing the boron concentration are not necessary during MODE 6 operations, isolation of all unborated water sources is an appropriate means of preventing unacceptable consequences of a boron dilution accident.
APPLICABLE SAFETY ANALYSES	The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6. The RCS boron concentration in MODE 6 satisfies Criterion 2 of the NRC Interim Policy Statement.
LCO	This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent inadvertent boron dilution during MODE 6 and, thus, avoid a reduction in SHUTDOWN MARGIN. (continued)

(continued)

BASES (continued)

LCO (continued)

This LCO does not rely on the OPERABILITY of systems or components. Rather, all of the following isolation valves associated with the potential flow paths of unborated water to the RCS must be secured in the closed position.

[For this facility, all potential unborated water sources and their associated isolation valves required to be secured in the closed position to prevent flow of unborated water into the RCS during MODE 6 are as follows:]

[For this facility, the following support systems are required OPERABLE to ensure unborated water sources isolation valve OPERABILITY:]

[For this facility, those required support systems which upon their failure do not declare the unborated water source isolation valves inoperable and their justification are as follows:]

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated.

ACTIONS

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "Immediately" for performance of Required Action A.1, shall not preclude completion of actions to establish a safe condition.

Condition A has been modified by a Note to require that Required Action A.3 must be completed whenever Condition A is entered. Also, a Note has been included to provide

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BASES (continued)

ACTIONS (continued)

clarification that each valve is treated as an independent entity for this LCO, with an independent Completion Time.

A.2

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valve(s) are secured in the closed position. [For this facility, secured constitutes the following:]

A.3

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to verify that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This surveillance involves verification that the valves are closed through a system walkdown. The 31-day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

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BASES (continued)

REFERENCES

1. [Unit Name] FSAR, Section [ ], "[Uncontrolled Boron Dilution]."
2. NUREG-0800, Standard Review Plan, Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the RCS."

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used 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 installed source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas-filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers 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.

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 with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. The need for a safety analysis for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2.

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

LCO

This LCO requires that two source range neutron flux monitors must be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. The OPERABILITY of the source range neutron flux monitors is established via a CHANNEL CHECK and ANALOG

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(continued)

BASES (continued)

LCO (continued)

CHANNEL OPERATIONAL TEST. OPERABILITY of the monitors also constitutes a separate continuous visual indication in the control room and an audible alarm in both the control room and the containment for each instrumentation train. The presence of an audible (count rate) signal in the control room and in the containment provides the operators with a method of quickly identifying significant changes in the source range neutron flux level.

[For this facility, the following support systems are required OPERABLE to ensure source range neutron flux monitor OPERABILITY:]

[For this facility, those required support systems which upon their failure do not declare the source range neutron flux monitors inoperable and their justification are as follows:]

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System Instrumentation."

ACTIONS

A.1 and A.2

With only one 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 actions to establish a safe condition.

A.3

With only one source range neutron flux monitor OPERABLE, action shall be initiated to restore the inoperable monitor to OPERABLE status within 7 days. Seven days is a

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BASES (continued)

ACTIONS (continued)

reasonable time in which corrective actions must be initiated considering the 72-hour boron sampling Frequency of SR 3.9.1.1, the suspension of CORE ALTERATIONS per Required Action A.1, and positive reactivity changes per Required Action A.2 above. Corrective actions, once started, must be continued until the monitor is restored to OPERABLE status.

B.1

With no source range neutron flux monitor OPERABLE, actions to restore a monitor to OPERABLE status shall be initiated within 15 minutes. Once initiated, actions shall be continued until a source range neutron flux monitor is restored to OPERABLE status. The 15-minute Completion Time is allowed for an operator to initiate corrective action.

B.2

With no 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 verify that the required boron concentration exists.

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

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is the comparison of the indicated parameter values for each of the functions. It is based on the assumption that the two

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 3.3.1, "Reactor Trip System Instrumentation."

SR 3.9.3.2

The performance of an ANALOG CHANNEL OPERATIONAL TEST provides assurance that the analog process control equipment and trip setpoints are within limits. [For this facility, an ANALOG CHANNEL OPERATIONAL TEST constitutes the following:] The 7-day Frequency has been shown through operating experience to be a conservative interval considering operating history data for the setpoint drift, and is further justified because any malfunctions would be detected during the CHANNEL CHECK which is performed every 12 hours.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants:"

   GDC 13, "Instrumentation and Control,"

   GDC 26, "Reactivity Control System Redundancy and Capability,"

   GDC 28, "Reactivity Limits," and

   GDC 29, "Protection Against Anticipated Operational Occurrences."

2. [Unit Name] FSAR, Section [ ], "[Uncontrolled Boron Dilution]."

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

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, there is no potential for containment pressurization as a result of an accident; 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 fuel assemblies within containment with irradiated fuel in containment, the equipment hatch must be held in place by at least [4] 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 through 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

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BASES (continued)

BACKGROUND (continued)

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

The requirements for 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, a mini-purge 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 mini-purge penetrations can be opened intermittently but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a technical 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 by the ESFAS in accordance with LCO 3.3.2.

[The mini-purge system remains operational in MODE 6, and all four valves are also closed by the ESFAS.]

or

[The mini-purge system is not used in MODE 6. All four [8]-inch valves are secured in the closed position.]

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BASES (continued)

BACKGROUND (continued)	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, by a manual isolation valve, blind flange, or by an equivalent isolation method. 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.
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. 1). Fuel-handling accidents, analyzed in Reference 2, include dropping a single fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity 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, result in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan Section 15.7.4, Rev. 1 (Ref. 2), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for [Unit Name] offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). During refueling, the requirements for containment penetrations satisfy Criterion 3 of the NRC Interim Policy Statement.
LCO	This LCO limits the consequences of a fuel-handling accident in containment by limiting the potential escape paths for fission-product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and

(continued)

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BASES (continued)

LCO (continued)

exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO provide the assurance 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 that releases through the valves are terminated such that radiological doses are within the acceptance limit.

[For this facility, the following support systems are required OPERABLE to ensure containment penetration OPERABILITY:]

[For this facility, those required support systems which upon their failure do not declare the containment penetrations inoperable and their justification are as follows:]

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of fuel assemblies within containment, with irradiated fuel in 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, "Containment." In MODES 5 and 6, when CORE ALTERATIONS or movement of fuel assemblies within containment, with irradiated fuel in containment, 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.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the plant must

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BASES (continued)

ACTIONS (continued)

be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of fuel assemblies within containment. Performance of these actions shall not preclude completion of actions to establish a safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This surveillance verifies 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 verify that the valves are not blocked from closing. Also the surveillance will verify that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The surveillance is performed every 7 days during CORE ALTERATIONS or movement of fuel assemblies within containment, with irradiated fuel in 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 radioactivity within the containment will not result in a release of fission-product radioactivity to the environment.

SR 3.9.4.2

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 3.3.2, "Engineered Safety Features Actuation System Instrumentation," the Containment Purge Isolation Signal System requires a CHANNEL CHECK every 7 days and an ANALOG CHANNEL OPERATIONAL TEST every 31 days to ensure the channel OPERABILITY during

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 Inspection and 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.

REFERENCES

1. [Unit Name] FSAR, Section [ ], "[Design Basis Fuel Handling Accidents]."

2. NUREG-0800, Standard Review Plan, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Rev. 1, July 1981.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

BASES

BACKGROUND	The purposes of the RHR 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, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water (CCW) System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay-heat removal is manually accomplished from the control room. The heat-removal rate is adjusted by controlling the flow of reactor coolant through the RHR 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 RHR System.
APPLICABLE SAFETY ANALYSES	If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to 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 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. One train of the RHR 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 RHR pump for short durations, under the condition that the boron concentration is not diluted. To ensure that the coolant temperature remains < 200°F, short durations of pump de-energization would be repeated only a few time. This conditional de-energizing

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

of the RHR pump does not result in a challenge to the fission-product barrier.

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

LCO

Only one RHR loop is required for decay-heat removal in MODE 6 with water level $\geq$ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE because the volume of water above the reactor vessel flange provides backup decay-heat-removal capability. At least one RHR 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 RHR loop includes an RHR 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.

[For this facility, an RHR loop in operation constitutes the following:]

[For this facility, the following support systems are required OPERABLE to ensure RHR System OPERABILITY:]

[For this facility, those required support systems which upon their failure do not declare the RHR System inoperable and their justification are as follows:]

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BASES (continued)

LCO (continued)

[For this facility, the supported systems impacted by the inoperability of a RHR System and the justification of whether or not each supported system is declared inoperable are as follows:] It should be noted that LCO 3.9.5 may need to be augmented with additional Conditions, if it is determined that the RHR System provides support to other systems included in the technical specifications during this MODE of operation.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 2-hour period. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot-leg nozzles, and RCS-to-RHR 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.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6 with the water level ≥ 23 ft above the top of the reactor vessel flange to provide decay-heat removal. The 23-foot water level was selected because it corresponds to the 23-foot requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, "Reactor Coolant System," and in Section 3.5, "Emergency Core Cooling System." RHR loop requirements in MODE 6 when water level is < 23 ft are located in LCO 3.9.6, "Residual Heat Removal and Coolant Circulation— Low Water Level."

ACTIONS

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

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron

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BASES (continued)

ACTIONS (continued)

concentration than that contained in the RCS because all of unborated water sources are isolated. Therefore, this immediate action will verify that the unborated water sources continued to be isolated.

A.2

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

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, a Completion Time of 15 minutes is allowed for an operation to initiate corrective actions.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This surveillance verifies that the RHR loop is OPERABLE, 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 RHR System.

REFERENCES

1. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level

BASES

BACKGROUND	The purposes of the RHR 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, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water (CCW) System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay-heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR 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 RHR System.
APPLICABLE SAFETY ANALYSES	If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to the 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 the 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 will eventually challenge the integrity of the fuel cladding, which is a fission-product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge. Although the RHR System does not meet a specific criterion of the NRC Interim Policy Statement, it was identified in the Policy Statement as an important contributor to risk reduction. Therefore, the RHR system is retained as a technical specification.

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BASES (continued)

LCO

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

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop consists of an RHR 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.

[For this facility, an RHR loop in operation constitutes the following:]

[For this facility, the following support systems are required OPERABLE to ensure RHR System OPERABILITY:]

[For this facility, those required support systems which upon their failure do not declare the RHR System inoperable and their justification are as follows:]

[For this facility, the supported systems impacted by the inoperability of an RHR System and the justification of whether or not each supported system is declared inoperable are as follows:] It should be noted that LCO 3.9.6 may need to be augmented with additional Conditions, if it is determined that the RHR System provides support to other systems included in the technical specifications during this MODE of operation.

APPLICABILITY

Two RHR loops are required to be OPERABLE and one RHR loop must be in operation in MODE 6 with the water level < 23 ft above the top of the reactor vessel flange to provide decay-heat removal. Requirements for the RHR System

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

in other MODES are covered by LCOs in Section 3.4, "Reactor Coolant System," and in Section 3.5, "Emergency Core Cooling System." RHR loop requirements in MODE 6 when the water level is ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal and Coolant Circulation—High Water Level."

ACTIONS

A.1 and A.2

If one RHR loop is inoperable or not in operation, actions shall be initiated and continued until the RHR loop is restored to OPERABLE status and to operation, or until ≥ 23 ft of water level is established above the reactor vessel flange while maintaining the correct boron concentration. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, "Residual Heat Removal and Coolant Circulation—High Water Level," and only one RHR loop is required to be OPERABLE and in operation. A Completion Time of 15 minutes is allowed for an operator to initiate corrective actions

B.1

If no RHR loop is OPERABLE or in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS because all of the unborated water sources are isolated. Therefore, this action will verify that these water sources continue to be isolated.

B.2

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

With no RHR loop OPERABLE or in operation, alternate actions shall have been initiated within 15 minutes under Condition

(continued)

(continued)

' S (continued)

ACTIONS (continued)

to establish ≥ 23 ft of water above the top of the reactor vessel flange while maintaining the correct boron concentration. Furthermore, when the LCO cannot be fulfilled, alternate decay-heat removal methods, as specified in the plant's Abnormal and Emergency Operating Procedures, should be implemented. This includes the removal of decay heat using the charging or safety injection pumps through the Chemical and Volume Control System (CVCS) with consideration of the boron concentration. The method used to remove decay heat should be the most prudent as well as the safest choice, based on plant conditions. The choice could be different if the reactor vessel head is in place rather than removed.

In addition to Actions B.1 and B.2, procedures and administrative controls as recommended by Generic Letter No. 88-17, "Loss of Decay Heat Removal," assure additional actions to mitigate the consequences of loss of decay-heat removal. The attachment to Generic Letter No. 88-17 includes recommended expeditious actions such as procedures and administrative controls. Procedures and administrative controls reasonably assure that containment closure will be achieved prior to the time at which core uncovery could result from a loss of RHR coupled with an inability to initiate alternate cooling or addition of water to the RCS inventory. An additional recommendation is the provision of at least two available or OPERABLE means of adding inventory to the RCS in addition to pumps that are a part of the normal systems. Procedures for use of these systems during loss of RHR events should also be provided.

Enclosure 2 to Generic Letter 88-17, "Guidance for Meeting Generic Letter 88-17," describes recommended programmed enhancements to be completed in a longer term than the expeditious actions and includes a discussion of potential future effects on technical specifications.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This surveillance verifies that one RHR loop is OPERABLE and in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide

(continued)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

sufficient decay-heat removal capability, and to prevent thermal and boron stratification in the core. In addition, this surveillance verifies that the other RHR is OPERABLE. In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements. the Frequency of 12 hours is sufficient considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR system in the control room.

REFERENCES

1. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND	The movement of fuel assemblies within containment, with irradiated fuel in containment, requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling cavity, refueling canal, fuel transfer canal, and spent fuel pool. Sufficient water is necessary to retain iodine fission-product activity in the water in the event of a fuel-handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.
APPLICABLE SAFETY ANALYSES	During movement of fuel assemblies, the water level in the refueling cavity and the refueling canal is an initial condition design parameter in the analysis of a fuel-handling accident in containment as postulated by NRC 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 100 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. 4). "Refueling Cavity Water Level" satisfies Criterion 2 of the NRC Interim Policy Statement.

(continued)

BASES (continued)

LCO

A minimum refueling cavity water level of 23 ft above the irradiated fuel is required to ensure that the radiological consequences of a postulated fuel-handling accident inside containment are within acceptable limits as provided by the guidance of Reference 3.

[For this facility, the following support systems are required OPERABLE to ensure refueling cavity water level OPERABILITY:]

[For this facility, those required support systems which upon their failure do not declare the refueling cavity water level inoperable and their justification are as follows:]

APPLICABILITY

Within the containment, LCO 3.9.7, "Refueling Cavity Water Level," is applicable when moving fuel assemblies in the presence of irradiated fuel assemblies. 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 3.7.11, "Fuel Storage Pool Water Level."

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of fuel assemblies shall be suspended immediately to ensure that a fuel-handling accident cannot occur. The suspension of fuel movement shall not preclude completion of movement to a safe position.

In the event that the required refueling cavity water level channels are found inoperable, the refueling cavity water level is considered to be not within limits and Required Action A.1 applies.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

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

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

REFERENCES

1. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel-Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, March 23, 1972.

2. [Unit Name] FSAR, Section [ ], "[Title]."

3. NUREG-0800, "Standard Review Plan," Section 15.7.4, "Radiological Consequences of Fuel-Handling Accidents," U.S. Nuclear Regulatory Commission.

4. Title 10, Code of Federal Regulations, Part 20.101(a), "Radiation Dose Standards for Individuals in Restricted Areas."

APPENDIX A

Acronyms

The following acronyms are used, but not defined, in the Standard Technical Specifications:

AC	alternating current
CFR	Code of Federal Regulations
DC	direct current
FSAR	Final Safety Analysis Report
LCO	Limiting Condition for Operation
SR	Surveillance Requirement
GDC	General Design Criteria or General Design Criterion

The following acronyms are used, with definitions, in the Standard Technical Specifications:

ACOT	ANALOG CHANNEL OPERATIONAL TEST
ADS	Automatic Depressurization System
ADV	atmospheric dump valve
AFD	AXIAL FLUX DIFFERENCE
AFW	auxiliary feedwater
AIRP	air intake, recirculation, and purification
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AOT	allowed outage time
APD	axial power distribution
APLHGR	average planar linear heat generation rate
APRM	average power range monitor
APSR	axial power shaping rod
ARO	all rods out
ARC	auxiliary relay cabinets
ARS	Air Return System
ARTS	Anticipatory Reactor Trip System
ASGT	asymmetric steam generator transient
ASGTPTF	asymmetric steam generator transient protective trip function
ASI	axial shape index
ASME	American Society of Mechanical Engineers

(continued)

ASTM	American Society for Testing Materials
ATWS	anticipated transient without scram
ATWS-RPT	anticipated transient without scram recirculation pump trip
AVV	atmospheric vent valve
BAST	boric acid storage tank
BAT	boric acid tank
BDPS	Boron Dilution Protection System
BIST	boron injection surge tank
BIT	boron injection tank
BOC	beginning of cycle
BOP	balance of plant
BPWS	banked position withdrawal sequence
BWST	borated water storage tank
BTP	Branch Technical Position
CAD	containment atmosphere dilution
CAOC	constant axial offset control
CAS	Chemical Addition System
CCAS	containment cooling actuation signal
CCGC	containment combustible gas control
CCW	component cooling water
CEA	control element assembly
CEAC	control element assembly calculator
CEDM	control element drive mechanism
CFT	core flood tank
CIAS	containment isolation actuation signal
COLR	CORE OPERATING LIMITS REPORT
COLSS	Core Operating Limits Supervisory System
CPC	core protection calculator
CPR	critical power ratio
CRA	control rod assembly
CRD	control rod drive
CRDA	control rod drop accident
CRDM	control rod drive mechanism
CREHVAC	Control Room Emergency Air Temperature Control System
CREFS	Control Room Emergency Filtration System
CREVS	Control Room Emergency Ventilation System
CRFAS	Control Room Fresh Air System
CS	core spray
CSAS	containment spray actuation signal

(continued)

APPENDIX A (continued)

CST	condensate storage tank
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DBE	Design Basis Event
DF	decontamination factor
DG	diesel generator
DIV	drywell isolation valve
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOP	dioctyl phthalate
DPIV	drywell purge isolation valve
DRPI	digital rod position indicator
EAB	exclusion area boundary
ECCS	Emergency Core Cooling System
ECW	essential chilled water
ECP	estimated critical position
EDG	emergency diesel generator
EFAS	Emergency Feedwater Actuation System
EFIC	emergency feedwater initiation and control
EFCV	excess flow check valve
EFPDs	effective full power days
EFPYs	effective full power years
EFW	emergency feedwater
EHC	electro-hydraulic control
EOC	end of cycle
EOC-RPT	end of cycle recirculation pump trip
ESF	engineered safety feature
ESFAS	Engineered Safety Feature Actuation System
ESW	essential service water
EVS	Emergency Ventilation System
FBACS	Fuel Building Air Cleanup System
FCV	flow control valve
FHAVS	Fuel Handling Area Ventilation System
FSPVS	Fuel Storage Pool Ventilation System
FRC	fractional relief capacity
FR	Federal Register
FTC	fuel temperature coefficient
FWLB	feedwater line break

(continued)

HCS	Hydrogen Control System; Hydrazine Control System
HCU	hydraulic control unit
HIS	Hydrogen Ignition System
HELB	high energy line break
HEPA	high efficiency particulate air
HMS	Hydrogen Mixing System
HPCI	high pressure coolant injection
HPCS	high pressure core spray
HPI	high pressure injection
HPSI	high pressure safety injection
HPSP	high power setpoint
HVAC	heating, ventilation, and air conditioning
HZP	hot zero power
ICS	Iodine Cleanup System
IEEE	Institute of Electrical and Electronic Engineers
IGSCC	intergranular stress corrosion cracking
IRM	intermediate range monitor
ISLH	inservice leak and hydrostatic
ITC	isothermal temperature coefficient
K-relay	control relay
LCS	Leakage Control System
LEFM	linear elastic fracture mechanics
LER	Licensee Event Report
LHGR	linear heat generation rate
LHR	linear heat rate
LLS	low-low set
LOCA	loss-of-coolant accident
LOCV	loss of condenser vacuum
LOMFW	loss of main feedwater
LOP	loss of power
LOPS	loss of power start
LOVS	loss of voltage start
LPCI	low pressure coolant injection
LPCS	low pressure core spray
LPD	local power density
LPI	low pressure injection
LPRM	local power range monitor
LPSI	low pressure safety injection
LPSP	low power setpoint

(continued)

APPENDIX A (continued)

LPZ	low population zone
LSSS	limiting safety system settings
LTA	lead test assembly
LTOP	low temperature overpressure protection
MAPLHGR	maximum average planar linear heat generation rate
MAPFAC	MAPLHGR factor
$MAPFAC_f$	MAPLHGR factor, flow-dependent component
$MAPFAC_p$	MAPLHGR factor, power-dependent component
MCPR	minimum critical power ratio
MCR	main control room
MCREC	main control room environmental control
MFI	minimum flow interlock
MFIV	main feedwater isolation valve
MFLPD	maximum fraction of limiting power density
MFRV	main feedwater regulation valve
MFW	main feedwater
MG	motor generator
MOC	middle of cycle
MSIS	main steam isolation signal
MSIV	main steam isolation valve
MSLB	main steam line break
MSSV	main steam safety valve
MTC	moderator temperature coefficient
NDT	nil-ductility temperature
NDTT	nil-ductility transition temperature
NI	nuclear instrument
NIS	Nuclear Instrumentation System
NMS	Neutron Monitoring System
NPSH	net positive suction head
NSSS	Nuclear Steam Supply System
ODCM	Offsite Dose Calculation Manual
OPDRV	operation with a potential for draining the reactor vessel
OTSG	once-through steam generator
PAM	post-accident monitoring
PCCGC	primary containment combustible gas control
PCI	primary containment isolation

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PCIV	primary containment isolation valve
PCHRS	Primary Containment Hydrogen Recombiner System
PCP	Process Control Program
PCPV	primary containment purge valve
PCT	peak cladding temperature
PDIL	power dependent insertion limit
PDL	power distribution limit
PF	position factor
PIP	position indication probe
PIV	pressure isolation valve
PORV	power operated relief valve
PPS	Plant Protective System
PRA	probabilistic risk assessment
PREACS	Pump Room Exhaust Air Cleanup System; Penetration Room Exhaust Air Cleanup System
PSW	plant service water
P/T	pressure and temperature
PTE	PHYSICS TESTS exception
PTLR	PRESSURE AND TEMPERATURE LIMITS REPORT
QA	quality assurance
QPT	quadrant power tilt
QPTR	QUADRANT POWER TILT RATIO
QS	quench spray
RACS	Rod Action Control System
RAOC	relaxed axial offset control
RAS	recirculation actuation signal
RB	reactor building
RBM	rod block monitor
RCCA	rod cluster control assembly
RCIC	reactor core isolation cooling
RCIS	Rod Control and Information System
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	Reactor Coolant System
REA	rod ejection accident
RHR	residual heat removal
RHRSW	residual heat removal service water
RMCS	Reactor Manual Control System
RPB	reactor pressure boundaries
RPC	rod pattern controller
RPCB	reactor power cutback

(continued)

APPENDIX A (continued)

RPIS	Rod Position Information System
RPS	Reactor Protection System
RPV	reactor pressure vessel
RS	recirculation spray
RT	reference temperature
RT_{NDT}	nil-ductility reference temperature
RTCB	reactor trip circuit breaker
RTD	resistance temperature detector
RTM	reactor trip module
RTP	RATED THERMAL POWER
RTS	Reactor Trip System
RWCU	reactor water cleanup
RWE	rod withdrawal error
RWL	rod withdrawal limiter
RWM	rod worth minimizer
RWP	Radiation Work Permit
RWST	refueling water storage tank
RWT	refueling water tank
SAFDL	specified acceptable fuel design limits
SBCS	Steam Bypass Control System
SBO	station blackout
SBVS	Shield Building Ventilation System
SCAT	spray chemical addition tank
SCI	secondary containment isolation
SCR	silicon controlled rectifier
SDV	scram discharge volume
SDM	SHUTDOWN MARGIN
SER	Safety Evaluation Report
SFRCS	Steam and Feedwater Rupture Control System
SG	steam generator
SGTR	steam generator tube rupture
SGTS	Standby Gas Treatment System
SI	safety injection
SIAS	safety injection actuation signal
SIS	safety injection signal
SIT	safety injection tank
SJAE	steam jet air ejector
SL	Safety Limit
SLB	steam line break
SLC	standby liquid control
SLCS	Standby Liquid Control System
SPMS	Suppression Pool Makeup System
SRM	source range monitor

(continued)

APPENDIX A (continued)

S/RV	safety/relief valve
S/RVDL	safety/relief valve discharge line
SSPS	Solid State Protection System
SSW	standby service water
SWS	Service Water System
STE	special test exception
STS	Standard Technical Specifications
TADOT	TRIP ACTUATING DEVICE OPERATIONAL TEST
TCV	turbine control valve
TIP	transversing incore probe
TLD	thermoluminescent dosimeter
TM/LP	thermal margin/low pressure
TS	Technical Specifications
TSV	turbine stop valve
UHS	Ultimate Heat Sink
VCT	volume control tank
VFTP	Ventilation Filter Testing Program
VHPT	variable high power trip
v/o	volume percent
VS	vendor specific
ZPMB	zero power mode bypass

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This draft report documents the results of the NRC staff review of new Standard Technical Specifications (STS) proposed by the Westinghouse Owners Group. The new STS were developed based on the criteria in the interim Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated February 6, 1987. The new STS will be used as bases for individual nuclear power plant owners to develop improved plant-specific technical specifications. The NRC staff is issuing this draft new STS for a 30 working-day comment period. Following the comment period, the NRC staff will analyze comments received, finalize the new STS, and issue them for plant-specific implementation. This report contains three volumes. Volume 1 contains the Specifications for all sections of the new STS. Volume 2 contains the Bases for Sections 2.0 - 3.3 of the new STS and Volume 3 contains the Bases for Sections 3.4 - 3.9 of the new STS.

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