NUREG–1432
Vol. 3

Standard Technical Specifications
Combustion Engineering 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 Combustion Engineering Owners Group (CEOG) 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 Combustion Engineering (CE). 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-1432, "Standard Technical Specifications, Combustion Engineering 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 rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady-state envelope. The limits placed on DNB-related parameters assure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO limits for minimum and maximum RCS pressures as measured at the pressurizer are consistent with operation within the nominal operating envelope and are bounded by those used as the initial pressures in the analyses.

The LCO limits for minimum and maximum RCS cold leg temperatures are consistent with operation at the indicated power level and are bounded by those used as the initial temperatures in the analyses.

The LCO limits for minimum and maximum RCS flows are bounded by those used as the initial flow rates in the analyses. The RCS flow rate is not expected to vary during plant operation with all pumps running.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ [1.3]. This is the acceptance limit for the RCS 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 struck control rod events. A key assumption for the

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	analysis of these events is that the core power distribution is within the limits of [LCO 3.1.7, "Regulating CEA Insertion Limits;" LCO 3.1.8, "Part-Length CEA Insertion Limits"; LCO 3.2.3, "AZIMUTHAL POWER TILT (T_q)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI) (Digital)]"; [LCO 3.1.7, "Regulating Rod Insertion Limits"; LCO 3.2.4, "AZIMUTHAL POWER TILT (T_q)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (Analog)]." The safety analyses are performed over the following range of initial values: RCS pressure [1785-2400] psia, core inlet temperature [500-580]°F, and reactor vessel inlet coolant flow rate [95-116]%. The RCS DNB parameter limits 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 analysis.
LCO	This LCO provides limits on the monitored process variables, pressurizer pressure, RCS cold leg temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient. The LCO numerical values for pressure, temperature, and flow are given for the measurement location, but have not been adjusted for instrument error. Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of this LCO.
APPLICABILITY	In MODE 1, the limits on RCS pressure, RCS cold leg temperature, and RCS flow rate must be maintained during steady-state operation in order to assure 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 so that DNBR is not a concern. The limit on RCS pressure may be exceeded during short-term operational transients such as a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER (RTP) per minute or a THERMAL POWER step increase of greater than 10% of RTP.

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BASES (continued)

APPLICABILITY (continued)

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% of RTP, increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB-related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core Safety Limits." Those limits are less restrictive than the limits of LCO 3.4.1, but violation of SLs merits a stricter, more severe Required Action. Should a violation of LCO 3.4.1 occur, the operator should check whether or not an SL may have been exceeded.

ACTIONS

A.1

Pressurizer pressure is a controllable and measurable parameter. With this parameter not within the LCO limits, action must be taken to restore the parameter.

The 2-hour Completion Time is based on plant operating experience that shows the parameter can be restored in this time period.

RCS flow rate is not a controllable parameter and is not expected to vary during steady-state operation. If the flow rate is not within the LCO limit, then power must be reduced, as required in 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 parameter provides sufficient time to adjust plant parameters, determine the cause of the off-normal condition, and restore the readings within limits. The Completion Time 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.

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BASES (continued)

ACTIONS (continued)

Pressurizer pressure and RCS 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 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 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

The 6 hours is a reasonable time that permits the plant power to be reduced at an orderly rate in conjunction with even control of steam generator (SG) heat removal.

C.1

Cold leg temperature is a controllable and measurable parameter. With this parameter not within the LCO limits, action must be taken to restore the parameter.

The 2-hour Completion Time is based on plant operating experience that shows that the parameter can be restored in this time period.

RCS cold leg temperature is considered out of limits if the equipment used to measure cold leg temperature is determined to be inoperable. Required Action C.1 applies to restoring such equipment to OPERABLE status.

D.1

If Required Action C.1 is not met within the associated Completion Time, THERMAL POWER must be reduced to $\leq$ 30% RTP. Plant operation may continue for an indefinite period of time in this condition. At the reduced power level, the potential for violat[illegible] of the DNB limits is greatly reduced.

The 6-hour Completi[illegible] is a reasonable time that permits power reduction at an orderly rate in conjunction with even control of SG 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 which 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 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 which are not within limits, the 12-hour Surveillance of cold leg temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady-state condition following load changes and other expected transient operations. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

[For this facility, RCS cold leg 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. This Surveillance verifies RCS flow within the bounds of the analyses. The 12-hour interval has been shown by operating experience to be sufficient to assess for potential degradation and to verify operation is within safety analysis assumptions.

This Surveillance is modified by a Note that only requires performance of this SR in MODE 1. The Note is necessary to allow measurement of RCS flow at normal operating conditions at power.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

[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 allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is within the bounds of the analyses.

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

The Surveillance is modified by a Note that states SR 3.0.4 is not applicable. The Note is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance cannot be performed in MODE 2 or below.

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

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for:

a. Operation within the existing instrumentation ranges and accuracies;

b. Operation within the bounds of the existing accident analyses; and

c. Operation with the reactor vessel above its minimum nil-ductility reference temperature when the reactor is critical.

The reactor coolant moderator temperature coefficient used in core operating and accident analysis is typically defined for the normal operating temperature range (532°F to 573°F). The Reactor Protection System receives inputs from the narrow range hot leg temperature detectors, which have a range of 520°F to 620°F. The average temperature (T_{avg}) is controlled using inputs of the same range. Nominal T_{avg} for making the reactor critical is 532°F. Safety and operating analyses for lower temperature have not been made.

APPLICABLE SAFETY ANALYSES

The low power safety analyses assume initial temperatures near the 520°F limit. (Ref. 1) These analyses for Design Basis Accidents (DBAs) 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.

This specification preserves limits used in the safety analysis and therefore satisfies Criterion 2 of the NRC Interim Policy Statement.

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BASES (continued)

LCO

The purpose of the LCO is to prevent criticality outside the normal operating regime (532°F to 573°F) and to prevent operation in an unanalyzed condition.

The LCO is only applicable below 535°F and provides a reasonable distance to the limit of 520°F. This allows adequate time to trend its approach and take corrective actions prior to exceeding the limit.

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODES 1 and 2 only, and in accordance with this specification. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2 when $K_{eff} \geq 1.0$. Coupled with the applicability definition for criticality is a temperature limit. Monitoring is required at and below a T_{avg} of 535°F. The no-load temperature of 544°F is maintained by the Steam Dump Control System.

[For this plant, if exceptions are taken to this LCO for performance of PHYSICS TESTS or other special tests, they will be specified herein.]

ACTIONS

A.1

If T_{avg} is below 525°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 sufficient 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 T_{avg} is determined to be inoperable. 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

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BASES (continued)

ACTIONS (continued)

readily and practically achieved in a 30-minute period. The allowed time reflects the urgency of maintaining the plant within the analyzed range and the ability of the plant to perform this action.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

T_{avg} is required to be verified above 520°F within 15 minutes prior to achieving criticality and every 30 minutes thereafter. The 15-minute time period allows the operator to adjust temperatures or delay criticality so that the LCO will not be violated. The 30-minute time is frequent enough to prevent inadvertent violation of the LCO.

While Surveillance is required whenever the reactor is critical and temperature is at or below 535°F, in practice the Surveillance is most appropriate during the period when the reactor is brought critical.

[For this facility, T_{avg} is measured as follows:]

REFERENCES

1. [Unit Name] FSAR, Section [15], "[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, and inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature. The heatup curve provides both heatup and criticality limits.

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 P/T 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 material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

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

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BASES (continued)

BACKGROUND (continued)

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 Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 3). Although any place 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), American Society for Testing Materials (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.

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

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BASES (continued)

BACKGROUND (continued)

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. Reactor coolant 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.

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

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BASES (continued)

BACKGROUND (continued)

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 includes the Reference 1 requirement that the limit be no less than 40°F above the heatup curve or the cooldown curve and not less than the minimum permissible temperature for the ISLH testing. However, the criticality limit 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 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 to determine the effect on the structural integrity of the RCPB components. The ASME Code Section XI, Appendix E (Ref. 7) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

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BASES (continued)

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. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

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 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.

d. Perform a LEFM analysis to establish the P/T 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 driven by postulating specific flaw sizes. Stress intensity factors for pressure and temperature are calculated and compared to a reference stress intensity factor.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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, 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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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 satisfy 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.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as

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BASES (continued)

LCO (continued)

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 pressure temperature regime or the severity of the rate of change of temperature;

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

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

APPLICABILITY

The RCS P/T limits specification provides a definition of acceptable operation for prevention of 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, their applicability is 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 Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

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BASES (continued)

APPLICABILITY (continued)

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.

ACTIONS

A.1 and A.2

Operation not within 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 pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. The evaluation must be completed, documented, and approved in accordance with established plant procedures and administrative controls.

ASME 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.

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(continued)

BASES (continued)

ACTIONS (continued)

Condition A is modified by a Note requiring both Required Action A.1 and Action A.2 completed whenever the condition is entered. The Note emphasizes the need to restore operation within limits and perform the evaluation 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 Actions A.1 and 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: a) the RCS remained in an unacceptable P/T region for an extended period of increased stress, or b) 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 Actions B.1 and B.2 must be implemented to reduce pressure and temperature.

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 Required Actions B.1 and B.2. A favorable evaluation must be completed, documented, and approved before returning to operating P/T conditions. However, if the favorable evaluation is accomplished while reducing P/T conditions, a return to power operation may be considered without completing Required Actions B.1 and B.2.

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

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BASES (continued)

ACTIONS (continued)

The 6-hour 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.

The 36-hour Completion Time for achieving MODE 5 also considers operating experience to reach the required MODE from full power 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. The pressure limit of 500 psig corresponds to the LCO 3.4.12 LTOP limit.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within LCO limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate-of-change limits are specified in hourly increments, 30 minutes 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 Surveillance requirement is given for criticality operations because LCO 3.4.2, "RCS Minimum Temperature for Criticality," contains a more restrictive requirement.

[For this facility, the combination of RCS pressure and temperature is 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."

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BASES (continued)

REFERENCES (continued)

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.

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. [NRC-approved topical report which defines the methodology for determining the P/T limits.]

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops--MODES 1 & 2

BASES

BACKGROUND

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

The secondary functions of the RCS include:

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

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

c. Carrying the soluble neutron poison, boric acid;

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

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

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains a SG and two reactor coolant pumps (RCPs). A RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This specification requires two RCS loops with both RCPs in operation in each loop. The intent of the specification is to require core heat removal with forced flow during power operation. Specifying two RCS loops provides the minimum necessary paths (two SGs) for heat removal.

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BASES (continued)

APPLICABLE SAFETY ANALYSES	Safety analyses contain various assumptions for the Design Bases Accident (DBA) initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate which is represented by the number of RCS loops in service.

Both transient and steady-state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming four RCPs 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 of most importance to RCP operation are the four pump coastdown, single pump locked rotor, single pump [broken shaft or coastdown], and rod withdrawal events (Ref. 1).

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

The number of loops and the associated RCS flow as represented by the number of pumps in operation satisfies Criterion 2 of the NRC Interim Policy Statement because the flow 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 adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG which is OPERABLE 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. [These specific instrumentation channels are:] SG and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the Reactor Protection System (RPS) in MODES 1 and 2. A reactor trip places the plant in MODE 3 if any SG level is ≤ [25]% as sensed by the RPS. The minimum water level to declare the SG OPERABLE is [25]%.

[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).

(continued)

BASES (continued)

ACTIONS

A.1

If the required number of RCS loops is not in operation, the Required Action is to reduce power and bring the plant to This MODE 3 lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits. It should be noted that the reactor will trip and place the plant in MODE 3 as soon as the RPS senses less than four RCPs operating.

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 of the required number of loops in operation and reactor coolant circulation every 12 hours. Verification includes flow rate and temperature monitoring, which help to ensure that forced flow is providing heat removal while maintaining the margin to DNB. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.4.2

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	indication of SG operation is the RCS water inventory balance performed in accordance with the requirements of LCO 3.4.13, "RCS Operational LEAKAGE."
REFERENCES	1. [Unit Name] FSAR, Section [ ], "[Title]."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops—MODE 3

BASES

BACKGROUND

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

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to satisfy the single failure criterion. Only one RCP need be OPERABLE to declare the associated RCS loop OPERABLE.

Reactor coolant natural circulation is not normally used but it is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured.

APPLICABLE SAFETY ANALYSES

Analyses have shown that the rod withdrawal event from MODE 3 with one RCS loop in operation is bounded by the rod withdrawal initiated from MODE 2. The analysis of the 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.

(continued)

BASES (continued)

LCO

The purpose of this LCO is to require two RCS loops to be available for heat removal thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable (> 25% water level) of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation.

The LCO Note permits a limited period of operation without RCPs. All RCPs may be de-energized for ≤ 1 hour per 8-hour period. This means that natural circulation has been established. When in natural circulation, boron reduction is prohibited because an even concentration distribution throughout the RCS cannot be assured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

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

An OPERABLE loop consists of at least one RCP providing forced flow for heat transport and an SG which is OPERABLE 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. [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, the following support systems are required to be OPERABLE to ensure RCS loop OPERABILITY in MODE 3:]

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BASES (continued)

LCO (continued)

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

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 forced flow heat removal has been lost.

The Required Action is restoration of the RCS loop to OPERABLE status within a Completion Time of 72 hours. This time allowance is a justified period to be without the redundant 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.

B.1

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

C.1 and C.2

If no loop is in operation, except as provided in Note 1 in the LCO section, all operations involving a reduction of RCS

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BASES (continued)

ACTIONS (continued)

boron concentration must be immediately suspended. This is necessary because boron dilution requires forced circulation for proper homogenization. Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This Surveillance requires verification of the required number of RCS loops in operation and reactor coolant circulation 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 departure from nucleate boiling. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.5.2

This Surveillance requires verification of water level in each SG is $\geq$ 25% every 12 hours. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within the safety analyses assumptions.

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

SR 3.4.5.3

Verification that the required number of RCPs are OPERABLE ensures that single failure criterion is met and that an additional RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	RCPs. 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	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 RCS loops is the removal of decay heat and transfer of this heat to the steam generator(s) (SG(s)) or shutdown cooling (SDC) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either reactor coolant pumps (RCPs) or SDC trains can be used for coolant circulation. The intent of this LCO is to provide forced flow from at least one RCP or one SDC train for decay heat removal and transport. The flow provided by one RCP loop or SDC train 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 loops and SDC trains 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.

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

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BASES (continued)

LCO

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

The LCO Note 1 permits all RCPs and SDC pumps to be stopped for 1 hour per 8-hour period. This means that natural circulation has been established using the SGs. With the RCS in natural circulation, boron reduction is prohibited because an even concentration distribution throughout the RCS cannot be assured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the RCS without the RCPs or SDC pumps depends on the core decay heat load and the length of time that the pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) limits or low temperature overpressurize protection (LTOP) limit) must be observed and forced SDC flow or heat removal via the steam generators must be reestablished prior to reaching the pressure limit. The circumstances for stopping both RCPs or SDC pumps are to be limited to:

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

b. Situations where an alternate heat removal path(s) through the SG(s) is in operation.

The LCO Note prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be assured.

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BASES (continued)

LCO (continued)

The LCO Note 2 requires that either of the following two conditions be satisfied before an RCP may be started with any RCS cold leg temperature ≤ 285°F:

a. Pressurizer water level must be < 35%; or

b. Secondary water temperature in each SG must be < 100°F above each of the RCS cold leg temperatures.

Satisfying either of the above conditions will preclude a large pressure surge in the RCS when the RCP is started.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program and 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. [These specific instrumentation channels are:]

Similarly, for the SDC System, an OPERABLE SDC train is composed of the OPERABLE SDC pump(s) providing forced flow to the SDC heat exchanger(s), along with appropriate flow and temperature instrumentation for control and indication. [These specific instrumentation channels are:] RCPs and SDC pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

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

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

APPLICABILITY

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

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BASES (continued)

APPLICABILITY (continued)

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 required RCS loop is OPERABLE and in operation, redundancy for heat removal is lost. The Required Action is to initiate activities to restore a second loop or train 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 only one required SDC train is OPERABLE and in operation, redundancy for heat removal is lost. The Required Action is to restore a second loop or train to OPERABLE status within 1 hour. The Completion Time 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 and an SDC train is OPERABLE, the plant must be placed in MODE 5 within the next 24 hours. Placing the plant in MODE 5 is a conservative action with regard to decay heat removal. With only one SDC train OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining SDC train, it would be safer to initiate that loss from MODE 5 ($\leq$ 200°F) rather than MODE 4 (200-300°F). The Completion Time of 25 hours is reasonable, based on operating experience, to reach MODE 5 from MODE 4, with only one SDC train operating, in an orderly manner and without challenging plant systems.

C.1 and C.2

If no RCS loops or SDC trains are OPERABLE or 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 loop or SDC train to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be

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BASES (continued)

ACTIONS (continued)

reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must continue until one loop or train is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

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

SR 3.4.6.2

This Surveillance requires verification of water level in the required SGs $\geq$ 25% every 12 hours. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.6.3

Verification that the required number of loops or trains are OPERABLE ensures that additional RCS loops or SDC trains 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 shutdown cooling (SDC) heat exchangers. While the principal means for decay heat removal (DHR) is via the SDC 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. In MODE 5 with RCS loops filled, the SDC trains are the principal means for heat removal. The number of trains in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC train for decay heat removal and transport. The flow provided by one SDC train 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 SDC train which must be OPERABLE and in operation. The second path can be another OPERABLE SDC train, or through the steam generators, each having an adequate water level.
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 SDC trains provide this circulation. (continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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 SDC trains do not meet any specific criterion of The NRC Interim Policy Statement, they were identified in the Policy Statement as an important contributor to risk reduction, and this LCO is thus retained as a Technical Specification.

LCO

The purpose of this LCO is to require that at least one of the SDC trains be OPERABLE and in operation with an additional SDC train OPERABLE or two SGs with secondary-side water level ≥ [25% wide range]. One SDC train provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second SDC train is normally maintained OPERABLE as a backup to the operating SDC train and satisfies the single failure criterion. However, if the standby SDC train is not OPERABLE, a sufficient alternate method of satisfying the single failure criterion is two SGs with their secondary-side water levels ≥ [25% wide range]. Should the operating SDC train fail, the SGs could be used to remove the decay heat.

The LCO Note 1 permits all SDC pumps to be stopped for up to 1 hour per 8-hour period. The circumstances for stopping both SDC trains are to be limited to:

a. Situations where pressure and temperature (P/T) increases can be maintained well within the allowable pressure (P/T and low temperature overpressure protection) and 10°F subcooling limits; or

b. Situations where an alternate heat removal path(s) through the SG(s) is in operation.

The LCO Note prohibits boron dilution when SDC forced flow is stopped because an even concentration distribution cannot be assured. Core outlet temperature is to be maintained at

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(continued)

BASES (continued)

LCO (continued)

least 10°F below saturation temperature so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the SG(s) can be used as a backup for SDC heat removal. To ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

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

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

The LCO Note 3 requires that either of the following two conditions be satisfied before a reactor coolant pump (RCP) may be started with any RCS cold leg temperature ≤ 285°F:

1. Pressurizer water level must be < 35%; or

2. Secondary water temperature in each SG must be < 100°F above each of the RCS cold leg temperatures.

Satisfying either of the above conditions will preclude a low temperature overpressure event due to a thermal transient when the RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of SDC trains from operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the heat removal function provided by the SDC trains.

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BASES (continued)

LCO (continued)

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger along with the appropriate flow and temperature instrumentation for control, protection, and indication. [These specific instrumentation channels are:]

SDC 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 SG Tube Surveillance Program.

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

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

APPLICABILITY

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

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

ACTIONS

A.1 and A.2

If only one SDC train is OPERABLE and in operation, and less than two SGs have secondary-side water levels ≥ [25% wide range], redundancy for heat removal is lost.

The Required Action is to initiate action to restore a second train to OPERABLE status or initiate action to restore the water level in the required SGs, and the action must be taken within 15 minutes. Either Required Action A.1 or A.2 will restore redundant decay heat removal paths. The

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BASES (continued)

ACTIONS (continued)

Completion Times of 15 minutes emphasize the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no train is OPERABLE or in operation, except as provided in Note 1 in the LCO, all operations involving the reduction of RCS boron concentration must be suspended and action to restore an SDC train to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing and the margin to criticality must not be reduced in this type of operation. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal. The action to restore must continue until one train is restored.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This Surveillance requires verification that at least one SDC train is 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 12-hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary-side water levels are ≥ [25% wide range] ensures that the single failure criterion is met if the second SDC train is not OPERABLE. The Note requires the Surveillance when the LCO requirement is being met by use of the SGs. If both SDC trains are OPERABLE, this Surveillance is not needed. The 12-hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.3

Verification that the second SDC train is OPERABLE ensures that the single failure criterion is met. The requirement also ensures that the additional train 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 3.4.8 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 and transfer of this heat to the shutdown cooling (SDC) 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, each facility shall expand this definition and background section to cover the mid-loop operation concerns expressed in GL 88-17, "Loss of Decay Heat Removal," 10 CFR 50.54(f). (Ref. 1)] The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid. In MODE 5 with loops not filled, only the SDC System can be used for coolant circulation. The number of trains in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC train 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 SDC trains provide this circulation. The flow provided by one SDC train 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 SDC trains do not meet any specific criterion of the NRC Interim Policy Statement, they were identified in the Policy

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	Statement as an important contributor to risk reduction, and this LCO is thus retained as a Technical Specification.

LCO

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

The LCO Note 1 permits the SDC pumps to be de-energized for ≤ 15 minutes when switching from one train to the other. The circumstances for stopping both SDC 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 SDC forced flow is stopped.

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

An OPERABLE SDC train is composed of an OPERABLE SDC pump providing forced flow to an OPERABLE SDC heat exchanger, along with the appropriate flow and temperature instrumentation for control, protection, and indication. [These specific instrumentation channels are:] SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

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

[For this facility, those required support systems which, upon their failure, do not require declaring the RCS loops

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BASES (continued)

LCO (continued)

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 SDC System.

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

ACTIONS

A.1

If only one required SDC train is OPERABLE and in operation, redundancy of heat removal is lost. The action is to initiate activities to restore a second train to OPERABLE status and the action must be taken within 15 minutes. The Completion Time emphasizes the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If both required SDC trains are inoperable or the required train is not in operation, except as provided by Note 1 in the LCO, the action requires immediate suspension of any operation for boron reduction and requires action to immediately start restoration of one train to OPERABLE status and operation. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal. The action to restore must continue until one train is restored.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This surveillance requires verification that at least one train is 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 departure from nucleate boiling. The 12-hour Frequency has been shown by

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.8.2

Verification that the required number of trains are OPERABLE ensures that the single failure criterion is met. The requirement also ensures that additional trains 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 SYSTEMS (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 volume, the heaters, and the heater control and power supplies. Pressurizer safety valves and pressurizer power-operated relief valves (PORVs) are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power-Operated Relief Valves (PORVs)," respectively.

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

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

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

The maximum level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus, both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus

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BASES (continued)

BACKGROUND (continued)

assuring that pressure relief devices PORVs or code safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design safety limit of 2750 psig or damage may occur to the PORV or pressurizer code safety valves.

The requirement to have two groups of pressurizer heaters assures that RCS pressure can be maintained. The pressurizer heaters maintain RCS pressure to keep the reactor coolant subcooled. Inability to control RCS pressure during natural circulation flow could result in a loss of single phase flow and a 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. No safety analyses are performed in lower MODES. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of 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 not exceeding the higher safety limit, the value for pressurizer level is nominal and is not adjusted for instrument error.

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B 3.4 REACTOR COOLANT SYSTEMS (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 volume, the heaters, and the heater control and power supplies. Pressurizer safety valves and pressurizer power-operated relief valves (PORVs) are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power-Operated Relief Valves (PORVs)," respectively.

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

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

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

The maximum level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus, both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus

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BASES (continued)

BACKGROUND (continued)

assuring that pressure relief devices PORVs or code safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design safety limit of 2750 psig or damage may occur to the PORV or pressurizer code safety valves.

The requirement to have two groups of pressurizer heaters assures that RCS pressure can be maintained. The pressurizer heaters maintain RCS pressure to keep the reactor coolant subcooled. Inability to control RCS pressure during natural circulation flow could result in a loss of single phase flow and a 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. No safety analyses are performed in lower MODES. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of 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 not exceeding the higher safety limit, the value for pressurizer level is nominal and is not adjusted for instrument error.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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 bases for the volume of reactor coolant released to the containment. The containment analysis performed using the mass and energy release demonstrated that the maximum resulting containment pressure was within design limits.

The above analyses are design basis analyses which are used to establish acceptance limits for the pressurizer. These design basis analyses are referenced to assess changes to the pressurizer to evaluate their effect on the acceptance limits.

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

The maximum pressurizer water level limit satisfies Criterion 2 of the NRC 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 code 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. 1), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water level <[35]% ensures that a steam bubble exists. Limiting the maximum operating water level preserves the

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BASES (continued)

LCO (continued)

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 subcooling margin to saturation 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 pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

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BASES (continued)

APPLICABILITY (continued)

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 gives 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 Shutdown Cooling System is inservice and therefore the LCO is not applicable.

ACTIONS

A.1 and A.2

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 plant to operation within the bounds of the safety analyses.

Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Further pressure and temperature reduction to MODE 4 places the plant into 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 for restoring this equipment to OPERABLE status.

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 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.

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BASES (continued)

ACTIONS (continued)

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 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 12 hours is reasonable, based on operating experience, to reach MODE 4 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This Surveillance ensures that during steady-state operation, pressurizer water level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing indicated level. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The Surveillance is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance). The Frequency of 92 days is considered adequate to detect heater degradation and has been shown by operating experience to be adequate.

SR 3.4.9.3

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	This surveillance demonstrates that the heaters can be manually transferred to, and energized by, emergency power supplies. The Frequency of 18 months is based on a typical fuel cycle and industry-accepted practice. This is consistent with similar verifications of emergency power.
REFERENCES	1. NUREG 0737, "Clarification of TMI Action Plan Requirements," November 1980. 2. Regulatory Guide 1.32, "Criteria for Safety Related Electrical 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 purpose of the two spring-loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the Reactor Protection System, two valves are used to assure that the Safety Limit of 2750 psia is not exceeded for analyzed transients during operation in MODES 1 and 2. Two safety valves are used for MODE 3 and portions of MODE 4. For the remainder of MODE 4, MODE 5 and MODE 6 with the head on, overpressure protection is provided by operating procedures and the LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." For these conditions, American Society of Mechanical Engineers (ASME) requirements are satisfied with one safety valve.

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

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

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME 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.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

All accident analyses in the FSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psia system design pressure plus 1%). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, loss of main feedwater, or main feedwater line break accident. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at less than 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME code.

Compliance with this specification is required to assure that the accident analysis and design basis calculations remain valid.

The pressurizer safety valves are components that are part of the primary success path and which function or actuate to 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, the pressurizer safety valves satisfy Criterion 3 of the NRC Interim Policy Statement.

The above analyses are design basis analyses which establish the acceptance limits for the pressurizer safety valves. Reference to the analyses for these design basis analyses is used to assess changes to the safety valves as they relate to the acceptance limits.

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psia) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure Safety Limit (SL), to maintain accident analysis assumptions, and to comply with ASME Code requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this specification is the reactor coolant pressure boundary (RCPB) SL of 110%

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BASES (continued)

LCO (continued)

of design pressure. Inoperability of one or both valves could result in exceeding the SL were a transient 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 condition. Only one valve will be removed from service at a time for testing. The [36]-hour exception is based on 18-hours outage time for each of the two valves. The 18-hour period is derived from operating experience that hot testing can be performed within 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 temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included although the listed accidents may not require both safety valves for protection.

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

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BASES (continued)

ACTIONS

A.1

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

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, the plant must be 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 is reasonable, based on operating experience, to reach MODE 3 from full power without challenging plant systems. Similarly, the 12 hours allowed is reasonable, based on operating experience, to reach MODE 4 without challenging plant systems. Below 275°F, overpressure protection is provided by LTOP. The change from MODES 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

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

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Vessels," 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 three devices for pressure relief functions: two safety valves which are safety-grade components and at least one PORV which is not a safety-grade device. The PORV is an electromagnetic pilot-operated valve that is automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. The PORV may also be manually operated using controls installed in the control room.

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

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

The PORV setpoint is at the high pressure reactor trip setpoint and below the opening setpoint for the pressurizer safety valves as required by Reference 2. The purpose of the relationship of these setpoints is to limit the number of transient pressure increase challenges which might open the PORV, which, if opened, could fail in the open position. The PORV setpoint thus limits the frequency of challenges from transients and limits the possibility of a small-break LOCA from a failed open PORV. Placing the setpoint below the pressurizer safety valve opening setpoint reduces the frequency of challenges to the safety valves which, unlike

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BASES (continued)

BACKGROUND (continued)	the PORV, cannot be isolated if they were to fail open. Accurate control of the PORV setpoint is therefore important for limiting the possibility of small-break LOCA. The primary purpose of this LCO is to ensure that the PORV, its setpoint, and the block valve are operating correctly so the potential for a small-break LOCA through the PORV pathway is minimized, or if a small-break LOCA were to occur through a failed-open PORV, the block valve could be manually operated to isolate the path. The PORV may also be manually operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available; a condition that may be encountered during loss of offsite power. Operators can manually open the PORVs to reduce RCS pressure in the event of a steam generator tube rupture with offsite power unavailable. The PORV may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater. The PORV functions as an automatic overpressure device and limits challenges to the safety valves. Although the PORV acts as an overpressure device for operational purposes, safety analyses [do not take credit for PORV actuation, but] do take credit for the safety valves. The PORV also provides low temperature overpressure protection (LTOP) during heatup and cooldown. LCO 3.4.12 addresses this function.
APPLICABLE SAFETY ANALYSES	The PORV small-break LOCA break size is bounded by the spectrum of piping breaks analyzed for plant licensing. Because the PORV small-break LOCA is located at the top of the pressurizer, the RCS response characteristics are different from RCS loop piping breaks; analyses have been performed to investigate these characteristics. The possibility of a small-break LOCA through the PORV is reduced when the PORV flow path is OPERABLE and the PORV

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

opening setpoint is established to be reasonably remote from expected transient challenges. The possibility is minimized if the flow path is isolated. The PORV opening setpoint has been established in accordance with Reference 2. It has been set so expected RCS pressure increases from anticipated transients will not challenge the PORV, minimizing the possibility of small-break LOCA through the PORV.

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

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

LCO

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

Valve OPERABILITY also means the PORV setpoint is correct. By ensuring that the PORV opening setpoint is correct, the PORV is not subject to frequent challenges from possible pressure increase transients, and therefore the possibility of a small-break LOCA through a failed-open PORV is not a frequent event.

[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:]

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BASES (continued)

APPLICABILITY

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

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, this 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 insurges 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 (LTOP System) addresses the PORV requirements in these MODES.

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

ACTIONS

A.1 and A.2

With the PORV inoperable and capable of being manually cycled, either the PORV must be restored or the flow path isolated within 1 hour. The block valve should be closed but power must be maintained to the block valve, since removal of power would render the block valve inoperable and Condition C would apply. Although the PORV may be designated inoperable, it may be able to be manually opened and closed and in this manner can be used 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

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BASES (continued)

ACTIONS (continued)

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 minor problems can be corrected or closure can be 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.

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

If one block valve is inoperable, then it must be restored to OPERABLE status, 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 Times of 1 hour are reasonable based on the small potential for challenges to the system during this time period and provide 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

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BASES (continued)

ACTIONS (continued)

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 Action cannot be met within the associated Completion Time, 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 within 12 hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Similarly, the 12 hours allowed is reasonable, based on operating experience, to reach MODE 4 from full power in an orderly manner and without challenging plant systems.

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 plant systems.

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BASES (continued)

ACTIONS (continued)

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 in 2 hours and the remaining block valve in 73 hours. The Completion Time of 1 hour to either restore the block valves or place the associated PORVs in manual control 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 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.

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 it can be closed if needed. The basis for the Frequency of 92 days is ASME XI

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

(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 interval (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 Action 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 at the RCS high 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 a typical refueling cycle and shutdown. The Frequency of 18 months is based on a typical refueling cycle and industry-accepted practice.

SR 3.4.11.3

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

SR 3.4.11.4

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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 the normal supply to the emergency supply and cycling the valves. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

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

2. U.S. Nuclear Regulatory Commission (NRC) Inspection and Enforcement (IE) Bulletin 79-05B, "Nuclear Incident at Three Mile Island," April 21, 1979.

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. LCO 3.4.3 provides the allowable combinations for operational P/T during cooldown, shutdown, and heatup to keep from violating 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 PTLR limits.

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

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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 HPSI pump is actuated by SI.

The LTOP System for pressure relief consists of two PORVs with reduced lift settings or an RCS vent of sufficient size. Two relief valves are required for redundancy. One PORV 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 actuation logic monitors RCS pressure and determines when the LTOP overpressure setting is approached. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The LCO 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 the LCO ensures the PTLR 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 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.

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

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BASES (continued)

BACKGROUND (continued)

capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the PTLR P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the specified flow capacity requires removing a pressurizer safety valve, removing a PORVs 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, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits during shutdown. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding [275]°F, the pressurizer safety valves prevent RCS pressure from exceeding the Reference 1 limits. At about [275]°F and below, overpressure prevention falls to the OPERABLE PORVs 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 PTLR 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 PTLR curves are revised, the LTOP System will be re-evaluated to ensure its functional requirements can still be satisfied using the PORV method or the depressurized and vented RCS condition.

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

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

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APPLICABLE SAFETY ANALYSES (continued)

Mass Input Type Transients

a. Inadvertent SI; or

b. Charging to letdown flow mismatch.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of residual heat removal cooling; or

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

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

a. Locking out all but [one] . SI pump, and all but [one] charging pump;

b. Immobilizing the SIT discharge isolation valves in their closed positions; and

c. Disallowing start of an RCP if secondary temperature is more than [46]°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 3 analyses demonstrate that either one PORV or the RCS vent can maintain RCS pressure below limits when only one HPSI pump and one charging pump are actuated by SI. Thus, the LCO requires only [one] HPSI pump and [one] charging pump OPERABLE during the LTOP MODES. Since neither the PORV nor the RCS vent can handle a full SI actuation, the LCO also requires the SITs isolated.

The isolated SITs must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of SIT discharge is

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 21 effective full power years of operation.

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. 4 and 5) requirements by having only [one] HPSI pump and [one] charging pump OPERABLE and SI actuation enabled for these pumps.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below [450] psig. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient of SI actuation of [one] HPSI pump and [one] charging pump. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing setpoints, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensure the Reference 1 limits will be met.

The PORV setpoints will be re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The PTLR P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement caused by neutron irradiation. Revised PTLR 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 represents the worst-case single active failure.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of [1.3] square inches is capable of mitigating the limiting allowed LTOP overpressure transient. In that event, this size vent maintains RCS pressures less than the minimum RCS pressure on the P/T limit curve.

The RCS vent size will also be re-evaluated for compliance each time the PTLR 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 only [one] HPSI pump and [one] charging pump OPERABLE and all SIT discharge isolation valves closed and immobilized. Specification 3.5.3, "ECCS—Shutdown," defines the pump OPERABILITY requirements. (Specification 3.3.2, "ESFAS Instrumentation," defines SI actuation OPERABILITY for the LTOP MODE 4 small-break LOCA, as discussed in the previous section.)

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

a. Two OPERABLE PORVs; or

b. A depressurized RCS and a RCS vent.

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set at [450] psig or less and testing

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BASES (continued)

LCO (continued)

has proven its ability to open at that setpoint, and motive power is available to the two valves and their control circuits. [For this facility, the power support systems for the PORV, its block valve, and their controls are as follows:]

An RCS vent is OPERABLE when open with an area of at least [1.3] 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:]

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.

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BASES (continued)

APPLICABILITY (continued)

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

ACTIONS

A.1 and B.1

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

The immediate Completion Time to initiate actions to restore restricted coolant input capability to the RCS reflects the maintaining overpressure protection of the RCS.

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

A SIT unisolated requires isolation within 1 hour. This is only required when the SIT pressure is at or more than the maximum RCS pressure for the existing temperature allowed in the PTLR.

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 12 hours. By increasing the RCS temperature to more than [175]°F, a SIT pressure of [600] psig cannot exceed the LTOP limits if the tanks are fully injected. Depressurizing the SIT below the LTOP limit stated in the PTLR also protects against such an event.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that 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 PORV inoperable, two PORVs must be restored to OPERABLE status within a Completion Time of 7 days. Two valves are required to meet the LCO requirement

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ACTIONS (continued)

and to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one PORV 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. 6). Thus, with one PORV 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 24-hour Completion Time to restore two PORVs OPERABLE in MODE 5 or in MODE 6 when the vessel head is on is reasonable to investigate and repair several types of PORV failures without exposure to a lengthy period with only one PORV OPERABLE to protect against overpressure events.

G.1

When (1) both PORVs are inoperable, or (2) a Required Action and an associated Completion Time of Condition A through Condition F are not met, or (3) the LTOP System is inoperable for any reason other than Condition A through Condition F, the RCS must be depressurized and a vent established within 8 hours. The vent must be sized at least [1.3] square inches to ensure the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action protects the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time of 8 hours to depressurize and vent the RCS is based on 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.

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BASES (continued)

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, only [one] HPSI pump and all but [one] charging pump are verified OPERABLE with the other pumps locked out with power removed and the SIT discharge isolation valves are verified closed and locked out.

The Frequency of 15 minutes before reducing RCS cold leg temperature below [275]°F ensures the maximum number of pumps permitted OPERABLE is not exceeded. The 12-hour interval considers operating practice to regularly assess potential degradation and verify operation within the safety analysis.

SR 3.4.12.4

The RCS vent at least [1.3] square inches is proven OPERABLE by verifying its open condition either:

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

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

The passive vent arrangement must only be open to be OPERABLE. This surveillance need only be performed if the vent is being used to satisfy the requirements of this LCO. The Frequencies consider operating experience with mispositioning of unlocked and locked valves, respectively.

SR 3.4.12.5

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 can be remotely verified open in the main control room.

The block valve is a remotely controlled motor-operated valve. The power to the valve motor operator is not required removed and the manual actuator 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 event.

The 72-hour Frequency considers operating experience with accidental movement of valves having remote control and position indication capabilities available where easily monitored. These considerations include the administrative controls over main control room access and equipment control.

SR 3.4.12.6

Performance of a CHANNEL FUNCTIONAL TEST is required within [12] hours of decreasing RCS temperature to at or below [275]°F and every 31 days to verify and, as necessary, adjust the PORV open setpoints. The CFT will verify on a monthly basis that the PORV lift setpoints are within the LCO limit. PORV actuation could depressurize the RCS and is not required.

The 12-hour Frequency considers the unlikelihood of a low temperature overpressure event during that time. The 31-day Frequency considers experience with equipment reliability.

A Note makes SR 3.0.4 not applicable. The test cannot be performed until the RCS is 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 3.4.12.7

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 LTOP range and accuracy to known input.

This Frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.

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BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50 (10 CFR 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. [Unit Name] FSAR, Section [15.], "[Title]."
4. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."
5. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."
6. 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,' pursuant to 10 CFR 50.44(f)."

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.

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BASES (continued)

BACKGROUND (continued)

This protection of LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss-of-coolant accident (LOCA).

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.

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BASES (continued)

LCO

a. No Pressure Boundary LEAKAGE

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

b. Unidentified LEAKAGE

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

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of identified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

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

Total primary-to-secondary LEAKAGE through all SGs 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 720 gallon per day (gpd) limit on one SG allocates the total 1 gpm allowed primary- to-secondary LEAKAGE equally between the two generators.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

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

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 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

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BASES (continued)

ACTIONS (continued)

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 RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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, & 3) define RCS PIVs as any two normally closed valves in series within the reactor coolant 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 permits 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 leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

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 condition 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. A

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BASES (continued)

BACKGROUND (continued)

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

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

a. Shutdown Cooling (SDC) System;
b. [Safety Injection System]; and
c. [Chemical and Volume Control System].

For this facility, 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 SDC 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 SDC System downstream of the PIVs from the RCS. Because the low pressure portion of the SDC System is typically designed for [600] psig, overpressurization failure of the SDC low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

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

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.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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. 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.

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 leak 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 leak 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). The observed

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BASES (continued)

LCO (continued)	rate is adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one-half power.
APPLICABILITY	In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for LOCA outside the containment.
ACTIONS	<u>A.1, A.2.1, and A.2.2</u> 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 RCPB. 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. 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. 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.

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BASES (continued)

ACTIONS (continued)

B.1 and B.2

If leakage cannot be reduced or the system isolated within the respective Completion Time, 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 without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve 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.

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 the valve.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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

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

SR 3.4.14.2 and SR 3.4.14.3

Verifying that the [SDC] autoclosure interlocks are operable ensures that RCS pressure will not pressurize the [SDC] 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 [SDC] design pressure will not be exceeded and the [SDC] relief valves will not lift. The frequency of [18 months] is based on engineering judgment and the fact that the testing of these interlocks is best performed during a refueling outage. This frequency has been shown to be acceptable through operating experience.

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 Leak Testing and Operational Changes," NUREG-0677, May 1980.

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BASES (continued)

REFERENCES (continued)

6. [ ]

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."

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 (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE is [(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 detection systems are included for monitoring both particulate and gaseous activities, because of their sensitivities and rapid responses to RCS LEAKAGE.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew-point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an

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

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

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 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 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

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BASES (continued)

ACTIONS (continued)

operation may continue while awaiting restoration of the containment air cooler condensate flow rate monitor to OPERABLE status.

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

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

instrument 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 it 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 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 Code of Federal Regulations, 10 CFR 100 (Ref. 1) specifies the maximum dose to the whole body and the thyroid an individual at the site boundary can receive for 2 hours during an accident. The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents. The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident. The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2-hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline 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 ensure 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 limits and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The analysis also assumes a reactor trip and a turbine trip at the same time as the SGTR event.

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

The safety analysis shows the radiological consequences of 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 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 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.

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 will be a small fraction of the allowed thyroid dose. The limit on gross specific

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(continued)

BASES (continued)

LCO (continued)

activity ensures the 2-hour whole body dose to an individual at the site boundary during the Design Basis Accident 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, 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 atmospheric dump valves and 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 required to reach MODE 3 from full power without challenging reactor emergency systems.

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BASES (continued)

ACTIONS (continued)

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 the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling must continue for trending. The DOSE EQUIVALENT I-131 must be restored to normal within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

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 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 reactor emergency 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 per 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 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.

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

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 the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

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BASES (continued)

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 SYSTEMS (RCS)

B 3.4.17 RCS Loops—Test Exceptions

BASES

BACKGROUND

This special test exception to LCO 3.4.4, "RCS Loops—MODES 1 and 2," and LCO 3.3.1, "RPS Instrumentation," permits reactor criticality under no flow conditions during PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) 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 (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 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 mixing 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

There are no transient or accident analyses which specify the allowed boundaries of this LCO. 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 a part of the LCO that they affect. This LCO was retained as a separate LCO for clarity.

LCO

This 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 this LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODES 1 and 2), and the appropriate tests could not be performed.

In MODE 2 where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no-flow conditions provided THERMAL POWER is <5% of RTP and the reactor trip setpoints of the OPERABLE power level channels are set at ≤ 20% RTP. These limits ensure no Safety Limits or fuel design limits will be violated.

The exemption is allowed even though there are no bounding safety analyses. These tests are allowed since they are performed under close supervision during the test program and provide valuable information on the plant's capability to cooldown without offsite power available to the reactor coolant pumps.

APPLICABILITY

This LCO ensures that the plant will not be operated in MODE 1 without forced circulation. It only allows testing under these conditions while in MODE 2. This testing establishes that heat input from nuclear heat does not exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.

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BASES (continued)

ACTIONS

A.1

If THERMAL POWER increases to > 5% RTP, the reactor must be tripped immediately. This ensures the plant is not placed in an unanalyzed condition, and prevents exceeding the specified acceptable fuel design limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

THERMAL POWER must be verified to be within limits once per hour to ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The hourly Frequency has been shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation within the LCO limits. 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 3.4.17.2

Within 12 hours of initiating PHYSICS TESTS, a CHANNEL FUNCTIONAL TEST must be performed on each logarithmic power level and linear power level neutron flux monitoring channel to verify OPERABILITY and adjust setpoints to proper values. This will ensure that the Reactor Protection System is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The interval is adequate to ensure that the appropriate equipment is OPERABLE prior to the tests to aid the monitoring and protection of the plant during these tests.

REFERENCES

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

2. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records," 1988.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Safety Injection Tanks (SITs)

BASES

BACKGROUND

The functions of the [four] SITs 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.

The refill phase of a LOCA follows immediately where 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 SITs inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer to establish a recovery level at the bottom of the core and ongoing reflood of the core with addition of safety injection (SI) water.

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

Each SIT is piped into one RCS cold leg via the injection lines utilized by the High Pressure Safety Injection and Low Pressure Safety Injection (HPSI and LPSI) systems. Each SIT is isolated from the RCS by a motor-operated isolation valve and two check valves in series. The motor-operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident.

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BASES (continued)

BACKGROUND (continued)

Additionally, the isolation valves are interlocked with the pressurizer pressure instrumentation channels to ensure that the valves will automatically open as RCS pressure increases above SIT pressure and to prevent inadvertent closure prior to an accident. The valves also receive a safety injection actuation signal (SIAS) to open. These features ensure that t' valves meet the requirements of the Institute of E. ..rical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the SITs will be available for injection without reliance on operator action.

The SIT gas and water volumes, gas pressure, and outlet pipe size are selected to allow three of the four SITs to partially recover the core before significant clad melting or zirconium-water reaction can occur following a LOCA. The need to ensure that three SITs are adequate for this function is consistent with the LOCA assumption that the entire contents of one SIT will be lost via the break during the blowdown phase of a LOCA.

APPLICABLE SAFETY ANALYSES

The SITs are taken credit for 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 SITs. Reference to the analyses for these DBAs is used to assess changes to the SITs, as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of SI flow. These assumptions include signal generation time, equipment starting times, and delivery time due to system piping. In the early stages of a LOCA with a loss of offsite power, the SITs provide the sole source of makeup water to the RCS. (The assumption of a loss of offsite power is required by regulations.) This is because the LPSI pumps, HPSI pumps, and charging pumps cannot deliver flow until the emergency diesel generators (EDGs) start, come to rated speed, and go through their timed loading sequence. In cold leg breaks, the entire contents of one SIT are assumed to be lost through the break during blowdown and the reflood phase.

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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

During this event, the SITs discharge to the RCS as soon as RCS pressure decreases to below SIT pressure. As a conservative estimate, no credit is taken for SI pump flow until the SITs are empty. This results in a minimum effective delay of over 60 seconds, during which the SITs must provide the core cooling function. The actual delay time does not exceed 30 seconds. No operator action is assumed during the blowdown stage of a large-break LOCA.

The worst-case small-break LOCA also assumes 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 SITs, with pumped flow then providing continued cooling. As break size decreases, the SITs and HPSI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the SITs continues to decrease until they are not required, and the HPSI pumps become solely responsible for terminating the temperature increase.

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

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

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

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

d. The core is maintained in a coolable geometry.

Since the SITs 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)

Since the SITs are passive components, single active failures are not applicable to their operation. The SIT isolation valves, however, are not single failure proof; therefore, whenever the valves are open, power is removed from their operators and the switch is key-locked open.

These precautions ensure that the SITs are available during an accident (Ref. 4). With power supplied to the valves, a single active failure could result in a valve closure, which would render one SIT unavailable for injection. If a second SIT is lost through the break, only two SITs would reach the core. Since the only active failure that could affect the SITs would be the closure of a motor-operated outlet valve, the requirement to remove power from these eliminates this failure mode.

The minimum volume requirement for the SITs ensures that three SITs can provide adequate inventory to reflood the core and downcomer following a LOCA. The downcomer then remains flooded until the HPSI and LPSI systems start to deliver flow.

The maximum volume limit is based on maintaining an adequate gas volume to ensure proper injection and the ability of the SITs to fully discharge, as well as limiting the maximum amount of boron inventory in the SITs.

A minimum of 25% narrow range level, corresponding to [1790] cubic feet of borated water, and a maximum of 75% narrow range level, corresponding to [1927] cubic feet of borated water, are used in the safety analyses as the volume in the SITs. To allow for instrument accuracy, a [28]% narrow range (corresponding to [1802] cubic feet) and a [72]% narrow range (corresponding to [1914] cubic feet) are specified. The analyses are based upon the cubic feet requirements; the percentage figures are provided for operator use because the level indicator provided in the control room is marked in percentages, not in cubic feet.

The minimum nitrogen cover pressure requirement ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analyses.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The maximum nitrogen cover pressure limit ensures that excessive amounts of gas will not be injected into the RCS after the SITs have emptied.

A minimum pressure of [593] psig and a maximum pressure of [632] psig are used in the analyses. To allow for instrument accuracy, a [615] psig minimum and [655] psig maximum are specified. The maximum allowable boron concentration of [2800] ppm is based upon boron precipitation limits in the core following a LOCA. Establishing a maximum limit for boron is necessary since the time at which boron precipitation would occur in the core following a LOCA is a function of break location, break size, the amount of boron injected into the core, and the point of ECCS injection. Post-LOCA emergency procedures directing the operator to establish simultaneous hot and cold leg injection are based on the worst-case minimum boron precipitation time. Maintaining the maximum SIT boron concentration within the upper limit ensures that the SITs do not invalidate this calculation. An excessive boron concentration in any of the borated water sources used for injection during a LOCA could result in boron precipitation earlier than predicted.

The minimum boron requirements of [1500] ppm are based on beginning of life reactivity values and are selected to ensure that the reactor will remain subcritical during the reflood stage of a large-break LOCA. During a large-break LOCA, all control element assemblies (CEAs) are assumed not to insert into the core, and the initial reactor shutdown is accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the SITs to prevent a return to criticality during reflood. Although this requirement is similar to the basis for the minimum boron concentration of the refueling water tank (RWT), the minimum SIT concentration is lower than that of the RWT since the SITs need not account for dilution by the RCS.

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

LCO

The LCO establishes the minimum conditions required to ensure that the SITs are available to accomplish their core

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BASES (continued)

LCO (continued)

cooling safety function following a LOCA. [Four] SITs are required to be OPERABLE to ensure that 100% of the contents of [three] of the SITs will reach the core during a LOCA.

This is consistent with the assumption that the contents of one tank spill through the break. If the contents of fewer than three tanks 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 SIT to be considered OPERABLE, the isolation valve must be fully open, with power removed, and the limits established in the SR 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 SIT OPERABILITY:]

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

APPLICABILITY

In MODES 1 and 2, and MODE 3 with RCS pressure ≥ 700 psia, the SIT OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the SITs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures ≥ 700 psia. Below 700 psia, 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 at pressures < 700 psia, and in MODES 4, 5, and 6, the SIT motor-operated isolation valves are closed to isolate the SITs from the RCS. This allows RCS cooldown and depressurization without discharging the SITs into the RCS or requiring depressurization of the SITs.

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

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BASES (continued)

ACTIONS

A.1

If the boron concentration of one SIT 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, but the reduced concentration effects on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the SIT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of three SITs, the consequences are less severe than they would be if an SIT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

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

B.1

If one SIT is inoperable, for a reason other than boron concentration, the SIT must be returned to OPERABLE status within 1 hour. In this condition, the required contents of three SITs cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1-hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action is taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the exposure of the plant to a LOCA in these conditions.

The SIT 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.

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BASES (continued)

ACTIONS (continued)

C.1 and C.2

If the SIT 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 by reducing pressurizer pressure to < 700 psia within 12 hours.

The allowed Completion Times are reasonable, based on operating experience related to the amount of 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 SIT 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 every 12 hours that each SIT isolation valve is fully open, as indicated in the control room, ensures that SITs 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 analysis assumptions. A 12-hour Frequency is considered reasonable in view of other 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 of each SIT nitrogen cover pressure and the borated water volume every 12 hours is sufficient to ensure adequate injection during a LOCA. Due to the static design of the SITs, a 12-hour Frequency usually allows the operator sufficient time to identify changes before the limits are reached. Operating experience has shown this Frequency to

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

be appropriate for early detection and correction of off-normal trends. In addition, alarms also signify off-normal conditions.

[For this facility, SIT borated water volume and nitrogen cover pressure are measured as follows:]

SR 3.5.1.4

The 31-day Frequency to verify that each SIT boron concentration is within the required limits was chosen because the static design of the SITs limits the means by which concentration could change, and the change by any of these means is slow. The 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 from the RCS has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the RWT, because the water contained in the RWT is within the SIT boron concentration requirements.

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

SR 3.5.1.5

Verification every 31 days that power is removed from each SIT isolation valve operator ensures that an active failure could not result in the undetected closure of an SIT motor-operated isolation valve. If this were to occur, only two SITs 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 the breaker is removed, a relatively easy surveillance, the 31-day Frequency was chosen to provide additional assurance 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 RCS pressure is < 2000 psia, thus allowing operational

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves. Should closure of the 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 [6.3], "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);

b. Rod ejection accident (REA);

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

d. Steam generator tube rupture (SGTR).

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

There are two phases of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the Reactor Coolant System (RCS) via the cold legs. After the blowdown stage of the LOCA stabilizes, injection flow is split equally between the hot and cold legs. After the refueling water tank (RWT) has been depleted, the ECCS recirculation phase is entered as the ECCS suction is automatically transferred to the containment sump.

Two redundant 100% capacity trains are provided. In MODES 1, 2, and 3, with pressurizer pressure ≥ 1700 psia, each train consists of high pressure safety injection (HPSI), low pressure safety injection (LPSI), and charging subsystems. In MODES 1, 2, and 3, with pressurizer pressure ≥ 1700 psia, both trains must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided in the event of a single active failure.

A suction header supplies water from the RWT or the containment sump to the ECCS pumps. Separate piping supplies each train. The discharge headers from each HPSI

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BASES (continued)

BACKGROUND (continued)

pump divide into four supply lines, each of which feeds the injection line to two RCS cold legs. Both HPSI trains feed into each of the four injection lines. The discharge header from each LPSI pump divides into two supply lines, each feeding the injection line to two RCS cold legs. Control valves or orifices are set to balance the flow to the RCS. This flow balance directs 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 initially depressurize the RCS below the shutoff head of the HPSI pumps, the charging pumps supply water to maintain inventory until the RCS pressure decreases below the HPSI pump shutoff head. During this period, the steam generators (SGs) must provide the core cooling function. The charging pumps take suction from the RWT on a safety injection actuation signal (SIAS) and discharge directly to the RCS through a common header. The normal supply source for the charging pumps is isolated on an SIAS to prevent non-condensible gas (e.g., air, nitrogen, or hydrogen) from being entrained in the charging pumps.

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

During a large-break LOCA, RCS pressure will decrease to < 200 psia in less than 20 seconds. The safety injection (SI) systems are actuated upon receipt of an SIAS. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the engineered safety feature (ESF) buses shed normal operating loads and are connected to the 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 safety injection tanks (SITs) and the RWT, covered in LCO 3.5.1,

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BASES (continued)

BACKGROUND (continued)	"Safety Injection Tanks (SITs)," and LCO 3.5.4, "Refueling Water Tank (RWT)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

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

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

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

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

d. Core is maintained in a coolable geometry; and

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

The LCO also limits the potential for a post-trip return to power following a steam line break (SLB) and ensures that containment temperature limits are met.

Both HPSI and LPSI subsystems are assumed to be OPERABLE in the large-break LOCA analysis at full power (Ref. 3). This analysis establishes a minimum required runout flow for the HPSI and LPSI pumps, as well as the maximum required response time for their actuation. The HPSI pumps and charging pumps are credited in the small-break LOCA analysis. This analysis establishes the flow and discharge head requirements at the design point for the HPSI pump. The SGTR and SLB analyses also credit the HPSI pumps, but are not limiting in their design.

The large-break LOCA event with a loss of offsite power and a single failure (disabling one ECCS train) establishes the OPERABILITY requirements for the ECCS. During the blowdown stage of a LOCA the RCS depressurizes as primary coolant

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control element assembly (CEA) insertion during small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

On smaller breaks, RCS pressure will stabilize at a value dependent upon break size, heat load, and injection flow. The smaller the break, the higher this equilibrium pressure. In all LOCA analyses, injection flow is not credited until RCS pressure drops below the shutoff head of the HPSI pumps.

The LCO ensures that an ECCS train will deliver sufficient water to match decay heat boiloff rates soon enough to minimize core uncovery for a large LOCA. It also ensures that the HPSI pump will deliver sufficient water during a small-break LOCA and provide sufficient boron to maintain the core subcritical following an SLB. For smaller LOCAs, the charging pumps deliver sufficient fluid to maintain RCS inventory until the RCS can be depressurized below the HPSI pumps' shutoff head. During this period of a small-break LOCA, the SGs continue to serve as the heat sink providing core cooling.

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

LCO

In MODES 1, 2, and 3, with pressurizer pressure ≥ 1700 psia, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available assuming there is 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 and 2, and in MODE 3 with pressurizer pressure ≥ 1700 psia, an ECCS train consists of an HPSI subsystem, an LPSI subsystem, and a charging pump.

Each train includes the piping, instruments, and controls to ensure the availability of an OPERABLE flow path capable of

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BASES (continued)

LCO (continued)

taking suction from the RWT on an SIAS and automatically transferring suction to the containment sump upon a recirculation actuation signal (RAS).

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the RWT to the RCS, via the HPSI and LPSI 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 part of its flow to the RCS hot legs via the shutdown cooling (SDC) suction nozzles. The charging pump flow path takes suction from the RWT and supplies the RCS via the normal charging lines.

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 and 2, and in MODE 3 with RCS pressure ≥ 1700 psia, the ECCS OPERABILITY requirements for the limiting Design Basis Accident (DBA) 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 HPSI pump performance is based on the small-break LOCA, which establishes the pump performance curve and has less dependence on power. The charging pump performance requirements are based on a small-break LOCA. The requirements of MODES 2 and 3, with RCS pressure ≥ 1700 psia, are bounded by the MODE 1 analysis.

The ECCS functional requirements of MODE 3 with RCS pressure < 1700 psia and MODE 4 are described in LCO 3.5.3.

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BASES (continued)

APPLICABILITY (continued)

As indicated in the Note, LCO 3.0.4 and SR 3.0.4 are expected 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 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 3.9.4, "Shutdown Cooling and Coolant Circulation—High Water Level," and LCO 3.9.5, "Shutdown Cooling and Coolant Circulation—Low Water Level."

ACTIONS

A.1

If one or more components are inoperable and at least 100% of the SI flow equivalent to a single OPERABLE ECCS train is available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72-hour Completion Time is based on an NRC Study (Ref. 4) using a reliability evaluation and is a reasonable amount time to effect many repairs.

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

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BASES (continued)

ACTIONS (continued)

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 OPERABLE equipment such that 100% of the SI flow equivalent to 100% of a single OPERABLE train remains available. This allows increased flexibility in plant operations when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 4) has shown that the impact with 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, for this LCO, all ECCS components 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:]

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

a. HPSI pump in train A and LPSI pump in train B (components serve different functions); and

b. Charging pump in train A and containment sump isolation valve in train B (components are in separate, parallel flow paths).

For example, unacceptable combinations of inoperable components include:

a. HPSI Pump in train A and HPSI control valve in train B failed closed (prevents HPSI flow to one cold leg);

b. HPSI pump in train A and containment sump isolation valve in Train B (removes both HPSI trains from service during recirculation); and

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BASES (continued)

ACTIONS (continued)

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

Reference 5 describes situations in which one component, such as a residual heat removal (RHR) crossover valve, can disable both ECCS trains. With one or more components inoperable, such that 100% of the equivalent flow to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable components cannot be returned to OPERABLE status within the associated Completion Times, 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 by reducing pressurizer pressure to < 1700 psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience related to the amount of 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 removing power or by key-locking the control in the correct position ensures that the valves cannot be inadvertently misaligned or change position as the result of an active failure. These valves are of the type described in Reference 5, which can disable the function of both ECCS trains and invalidate the accident analysis. A 12-hour Frequency is considered reasonable in view of other administrative controls that will ensure that a mispositioned valve is an unlikely possibility.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power-operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31-day Frequency of this SR was derived from inservice testing requirements, which specify performing valve testing at least once every 92 days. The Frequency is further justified in view of the procedural control governing valve operation and to provide added assurance of correct valve positions.

SR 3.5.2.3

With the exception of systems in operation, 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 a SIAS or during SDC. The 31-day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the adequacy of the procedural controls governing system operation.

SR 3.5.2.4

Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the American Society of Mechanical Engineers (ASME) Code. Because RCS pressure is higher than the discharge head of the HPSI and

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

LPSI pumps, they are tested on recirculation flow. Delivering their minimum recirculation flow, the pumps operate near their shutoff head. This test thus confirms one point on their design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY and trend performance, as well as detect incipient failures by indicating abnormal performance. A quarterly Frequency for such tests is a Code requirement.

SR 3.5.2.5

Discharge head at minimal recirculation flow is a normal test of pump performance required by Section XI of the ASME Code. A quarterly Frequency for such tests is a Code requirement. Such inservice inspections detect component degradation and incipient failures.

SR 3.5.2.6, SR 3.5.2.7, and SR 3.5.2.8

These SRs demonstrate that each automatic ECCS valve actuates to its required position on an actual or simulated SIAS and on a RAS, that each ECCS pump starts on receipt of an actual or simulated SIAS, and that the LPSI pumps stop on receipt of an actual or simulated RAS. The 18-month Frequency was developed considering 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 the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.9

Realignment of valves in the flow path on a SIAS is necessary for proper ECCS performance. The safety injection valves have stops to position them properly so that flow is restricted to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. This SR is not required for plants with flow-limiting orifices. The 18-month Frequency is based on the same factors as those stated above for SR 3.5.2.6, SR 3.5.2.7, and SR 3.5.2.8.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.10

Periodic inspection of the containment sump ensures that it is unrestricted and stays in proper operating condition. An 18-month Frequency was developed considering it is prudent that this surveillance only be performed during an outage. This is due to plant conditions needed to perform the surveillance and access to the location. This Frequency is sufficient to detect abnormal degradation and is confirmed by operating experience.

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 [6], "[Emergency Core Cooling System]."

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

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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS—Shutdown

BASES

BACKGROUND

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

In MODE 3 with pressurizer pressure < 1700 psia and in MODE 4, an ECCS train is defined as one high pressure safety injection (HPSI) subsystem. The HPSI flow path consists of piping, valves, and pumps that enable water from the refueling water tank (RWT) to be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

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

In these MODES, the decay heat generation and RCS blowdown rates are such that a single HPSI pump is capable of providing the core cooling function in the event of a loss-of-coolant accident (LOCA).

LCO

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

An additional relaxation in the ECCS requirements for these MODES is that only one HPSI subsystem is required. This requirement dictates that single failures are not considered during these MODES of operation.

In MODE 3 with pressurizer pressure < 1700 psia, an ECCS subsystem is composed of a single HPSI subsystem. Each

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BASES (continued)

LCO (continued)	HPSI subsystem includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT and transferring suction to the containment sump. During an event requiring ECCS actuation, a flow path is required to supply water from the RWT to the RCS via the HPSI 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. With RCS pressure < 1700 psia, one HPSI pump is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements. The low pressure safety injection (LPSI) pumps may therefore be released from the ECCS train for use in shutdown cooling (SDC). In MODE 4 with RCS cold leg temperature ≤ 285°F, a maximum of one HPSI pump is allowed to be OPERABLE in accordance with LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." [For this facility, the following support systems are required to be OPERABLE to ensure HPSI train OPERABILITY:] [For this facility, those required support systems which, upon their failure, do not require declaring the HPSI train inoperable and their justification are as follows:]
APPLICABILITY	In MODES 1, 2, and 3 with RCS pressure ≥ 1700 psia, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2, "ECCS—Operating." In MODE 3 with RCS pressure < 1700 psia and in MODE 4, one OPERABLE ECCS train is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements. In MODES 5 and 6, plant conditions are such that 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

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BASES (continued)

APPLICABILITY (continued)

requirements are addressed by LCO 3.9.4, "Shutdown Cooling and Coolant Circulation—High Water Level," and LCO 3.9.5, "Shutdown Cooling and Coolant Circulation—Low Water Level."

ACTIONS

A.1

For this facility, an OPERABLE ECCS HPSI train consists of a HPSI pump, a heat exchanger, piping, instruments, controls, cables, and other equipment to ensure an OPERABLE flow path.

If no HPSI pump is OPERABLE, the unit is not prepared to respond to a LOCA. The 1-hour Completion Time to restore at least one HPSI train to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the unit in MODE 5 where an ECCS train is not required.

The Note associated with Condition A is intended to convey that continuation of Actions is needed to restore the HPSI train to OPERABLE status in view of the fact that the plant cannot go to MODE 5 because there is no SDC train available.

B.1

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

The Note associated with Required Action B.1 is intended to restrict entry into this condition to those times when at least one SDC train is OPERABLE. The Note also is intended to convey that further action to reach MODE 5 should be suspended if, while in Condition B, all SDC trains become inoperable.

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BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.5.3.1 The applicable surveillance descriptions from Bases 3.5.2 apply.
REFERENCES	The applicable references from Bases 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Tank (RWT)

BASES

BACKGROUND

The RWT supports the ECCS and the Containment Spray System by providing a source of borated water for engineered safety feature (ESF) pump operation.

The RWT supplies two ECCS trains by separate, redundant supply headers. Each header also supplies one train of the Containment Spray System. A locked-open, motor-operated isolation valve is provided in each header to allow the operator to isolate the usable volume of the RWT from the ECCS after the ESF pump suction has been transferred to the containment sump following depletion of the RWT during a loss-of-coolant accident (LOCA). A separate header is used to supply the Chemical and Volume Control System (CVCS) from the RWT. Use of a single RWT to supply both trains of the ECCS is acceptable since the RWT is a passive component and passive failures are not assumed to occur coincidently with the Design Basis Event during the injection phase of an accident. Not all the water stored in the RWT is available for injection following a LOCA; the location of the ECCS suction piping in the RWT will result in some portion of the stored volume being unavailable.

The high pressure safety injection (HPSI), low pressure safety injection (LPSI), and containment spray pumps are provided with recirculation lines that ensure that each pump can maintain minimum flow requirements when operating at shutoff head conditions. These lines discharge back to the RWT, which vents to the atmosphere. When the suction for the HPSI and containment spray pumps is transferred to the containment sump, this flow path must be isolated to prevent a release of the containment sump contents to the RWT. If not isolated, this flow path could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ESF pumps.

This LCO ensures that:

a. The RWT contains sufficient borated water to support the ECCS during the injection phase;

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BASES (continued)

BACKGROUND (continued)

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

c. The reactor remains subcritical following a LOCA.

Insufficient water inventory in the RWT could result in insufficient cooling capacity of the ECCS 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 a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWT provides a source of borated water to the HPSI, LPSI, containment spray and charging 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," and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWT in order to evaluate their effects in relation to the acceptance limits.

The volume limit of [362,800] gallons is based on two factors:

a. Sufficient deliverable volume must be available to provide at least 20 minutes (plus a 10% margin) of full flow from all ESF pumps prior to reaching a low-level switchover to the containment sump for recirculation; and

b. The containment sump water volume must be sufficient to support continued ESF pump operation after the switchover to recirculation occurs. This sump volume water inventory is supplied by the RWT borated water inventory.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Twenty minutes is the point at which 75% of the design flow of one HPSI pump is capable of meeting or exceeding the decay heat boiloff rate.

When ESF pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head (NPSH) for the HPSI and containment spray pumps. The RWT capacity must be sufficient to supply this amount of water without considering the inventory added from the safety injection tanks or Reactor Coolant System (RCS), but accounting for loss of inventory to containment subcompartments and reservoirs due to containment spray operation and to areas outside containment due to leakage from ECCS injection and recirculation equipment.

The [1720] ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum level in the RWT, the reactor will remain subcritical in the cold condition following mixing of the RWT and RCS water volumes. Small-break LOCAs assume that all control rods are inserted, except for the control element assembly (CEA) of highest worth, which is withdrawn from the core. Large-break LOCAs assume that all CEAs remain withdrawn from the core. The most limiting case occurs at beginning of life.

The maximum boron limit of [2500] ppm in the RWT is based on boron precipitation in the core following a LOCA. With the reactor vessel at saturated conditions, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point will be reached where boron precipitation will occur in the core. Post-LOCA emergency procedures direct the operator to establish simultaneous hot and cold leg injection to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA. Boron concentrations in the RWT in excess of the limit could result in precipitation earlier than assumed in the analysis.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The upper limit of [100]°F and the lower limit of [40]°F on RWT temperature are the limits assumed in the accident analysis. Although RWT temperature affects the response of several analyses, the upper and lower limits established by the LCO are not limited by any of these analyses.

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

LCO

The RWT 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) and to cool and cover the core in the event of a LOCA, that the reactor remains subcritical following a DBA, and that adequate level exists in the containment sump to support ESF pump operation in the recirculation mode.

To be considered OPERABLE, the limits established in SR 3.5.4.1, SR 3.5.4.2, and SR 3.5.4.3 for water volume, boron concentration, and temperature must be met.

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

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

APPLICABILITY

In MODES 1, 2, 3, and 4, the RWT OPERABILITY requirements are dictated by the 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 RWT must be OPERABLE to support their operation.

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

(continued)

BASES (continued)

ACTIONS

A.1

If RWT borated water volume, boron concentration, or borated water temperature is not within limits, the RWT must be returned to within limits within 1 hour. In this condition, neither the ECCS nor Containment Spray System can perform their design functions. 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 these systems are not required. The short period of 1 hour to restore the RWT to OPERABLE is based on this condition simultaneously affecting multiple trains.

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

B.1 and B.2

If the RWT 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 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 3.5.4.1

Verification every 24 hours that the RWT borated water temperature is maintained within the limits assumed in the accident analysis is frequent 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

are within the operating temperature limits of the RWT. With ambient within this range, the RWT temperature should not exceed the limits.

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

SR 3.5.4.2

Verification every 7 days that RWT water volume is maintained above the required minimum level ensures that a sufficient initial supply is available for injection and to support continued ESF pump operation on recirculation. Since the RWT volume is normally stable and part of a passive subsystem and is provided with a low-level alarm, a 7-day Frequency is appropriate and has been shown to be acceptable through operating experience.

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

SR 3.5.4.3

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

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

REFERENCES

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment (Atmospheric)

BASES

BACKGROUND

The containment is comprised of the concrete reactor building (RB), its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA), 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 biological shielding from the fission products that may be present in the containment atmosphere following accident conditions.

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

The concrete RB is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage-limiting boundary of the containment. Maintaining the containment OPERABLE limits of 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 are in conformance with 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions.

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APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressure 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 main steam line break (MSLB), and a control element assembly ejection accident (Ref. 3). In addition, release of significant fission-product radioactivity within containment can occur from a LOCA. 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. 3). 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 maximum peak containment pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage-rate testing. For this unit, L_a = [0.1]% per day and P_a = [55.7] psig, resulting from the limiting design basis MSLB (Ref. 3).

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

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 dose 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. 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 containment LCOs support this LCO by ensuring that:

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

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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 this LCO will ensure a containment configuration that is structurally sound and will limit leakage to those leakage rates assumed in the safety analyses. 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. The 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." The SRs require either that the associated containment isolation valves 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 test required by SR 3.6.3.7 and Appendix J ensures 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.

   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 the appropriate SRs of LCO 3.6.3. 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.

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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. 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 provide assurance 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.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment 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 3.9.3, "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

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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining containment OPERABLE requires compliance with the visual examinations and leakage-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 except [42]-inch purge valves (Type C leakage tests). Leakage-rate testing of 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 3.02 (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)

SURVEILLANCE REQUIREMENTS (continued)

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 (Dual)

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 radioactive material that may be released from the reactor core following a Design Basis Accident (DBA), so 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 ellipsoidal bottom. It is 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.

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 to 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions.

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APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate, so 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 that exceed 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 main steam line break (MSLB), and a control element assembly (CEA) ejection accident (Ref. 3). In addition, release of significant fission-product radioactivity within containment can occur from a LOCA or a CEA ejection accident. The DBA analyses assume that the containment and shield building are OPERABLE at event initiation, so that, for 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 [0.50]% 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 maximum peak containment pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage-rate testing. For this unit, L_a = [0.50]% per day and P_a = [42.3] psig, which results from the limiting 75% THERMAL POWER MSLB (Ref.).

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

b. A member of the general public who remains at the low-population-zone boundary for the duration of the accident.

(continued)

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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 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 that:

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 Programs; 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 those LCOs and not in LCO 3.6.1.

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BASES (continued)

LCO (continued)

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. 2) 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." The SRs require either 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 test required by SR 3.6.3.7 and Appendix J ensures 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.

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 the appropriate SRs of LCO 3.6.3. 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.

b. The OPERABILITY of the containment equipment hatch is ensured by compliance with the leakage criteria established by 10 CFR 50, Appendix J (Ref. 2).

c. The OPERABILITY of the containment air locks 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),

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BASES (continued)

LCO

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 conform with those of 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 provide assurance 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 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event that containment is inoperable, it 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 OPERABLE during MODES 1, 2, 3, and

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(continued)

BASES (continued)

ACTIONS (continued)

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 cont nment 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 containment OPERABLE requires compliance with the visual examinations and leakage-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 leakage from 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 required by Appendix J or identified in the Containment Leakage Rate Testing Program. Thus, SR 3.0.2 (which allows Surveillance 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."

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BASES (continued)

REFERENCES (continued)

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 & Dual)

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES or operations.

Each air lock is nominally a right circular cylinder, 10 feet in diameter, with a door at both ends. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air-lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double-gasketed seals and local leakage-rate testing capability to ensure pressure integrity. To effect a leak-tight seal, the air-lock design uses pressure-seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

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

The containment air locks form part of the containment pressure boundary. As such, air-lock integrity and air tightness is essential to limit offsite doses from a DBA. Not maintaining air-lock integrity or air tightness may result in offsite doses in excess of those described in the plant safety analyses. All leakage rate requirements and SRs are in conformance with 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The containment air lock LCO is derived from the requirements related to the control of offsite radiation doses from major accidents by verifying that the actual containment leakage rate does not exceed the value assumed in the plant safety analyses. For example, the loss-of-cooling accident (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 dose 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 LOCA, a main steam line break, and a control element assembly ejection accident (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 0.10% of containment air weight per day (Ref. 4). 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 maximum peak containment 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.

(continued)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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 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 support containment OPERABILITY following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry and exit from containment.

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 exposure resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air-lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air-lock leakage test, and both air-lock doors must be OPERABLE. The interlock allows only one air-lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. 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.

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(continued)

BASES (continued)

LCO (continued)

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

[For this facility, those required support systems which, upon their failure, do not require declaring the 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 3.9.3, "Containment Building Penetrations."

The Required Actions for Conditions A, B, and 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 because of 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 included to provide clarification that for this LCO all containment air locks are treated as an entity with a single Completion Time.

(continued)

BASES (continued)

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 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 the use of an OPERABLE air-lock door. One of these two Required Actions must be completed within the 24-hour Completion Time. The associated 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 and closed OPERABLE air-lock door. This ensures that an acceptable containment leakage boundary is maintained. The leakage rate acceptance criteria are as defined in SR 3.6.2.1. The periodic interval of 31 days is based on engineering judgment and is considered adequate in view of other administrative controls, such as door status indicators available to the operator that 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 lock to ensure that only one door is opened at a time and that the opened door is immediately closed.

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BASES (continued)

ACTIONS (continued)

C.1 and C.2

With one or more air locks inoperable for reasons other than described in Condition A or B, one door in the containment air lock must be verified to be closed. This action must be completed within the 1-hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24-hour Completion Time. The specified time period is considered reasonable for restoring 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 acceptance criteria were established during initial air lock and containment OPERABILITY testing and in

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

accordance with 10 CFR 50, Appendix J, are stated in this SR. 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.

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 opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when containment is entered, this test is only required to be performed prior to entering containment, but is not required more frequently than every 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.

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BASES (continued)

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, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

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

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves (Atmospheric & 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 an automatic 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 the requirements of 10 CFR 50, Appendix A, 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 analysis. One of these barriers may be a closed system inside containment, in accordance with GDC 57.

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

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BASES (continued)

BACKGROUND (continued)

blind flanges) ensures containment OPERABILITY is maintained during accident conditions.

The OPERABILITY requirements for containment isolation valves help ensure that 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 accident analysis will not be exceeded.

The purge valves were designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor RCS leakage prior to personnel entry into containment. There are two sets of purge valves: [42]-inch normal purge and exhaust valves and [8]-inch mini-purge and exhaust valves. The normal and mini-purge supply and exhaust lines are each supplied with inside and outside containment isolation valves but share common [42]-inch supply and exhaust penetration lines.

The normal purge valves are designed for purging the containment atmosphere to the plant stack while introducing filtered makeup from the outside to provide adequate ventilation for personnel comfort when the plant is shut down during refueling operations and maintenance. Motor-operated isolation valves are provided inside the containment and air-operated isolation valves are provided outside the containment. The valves are operated manually from the control room. The valves will close automatically upon receipt of a containment purge isolation signal (CPIS). The air-operated valves fail closed upon a loss of air. 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.

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BASES (continued)

BACKGROUND (continued)

The mini-purge valves are designed for continuous purging of the containment during power operation to allow operator access. Both the inside and outside isolation valves are air operated and fail closed upon a loss of air. The valves automatically close upon receipt of a safety injection actuation signal, a containment isolation actuation signal, or a CPIS.

Open normal purge valves, or a failure of the mini-purge valves to close, following an accident that releases contamination to the containment atmosphere would cause a significant increase in the offsite radiation dose. This could result in exceeding the dose limits of Reference 2.

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 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 leakage rate from the containment. This LCO is intended to ensure that the offsite dose limits are not exceeded (actual containment leakage rate does not exceed the value assumed in the safety analysis). As part of the containment boundary, containment isolation valve and containment purge valve OPERABILITY are essential to containment OPERABILITY. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss-of-coolant accident (LOCA), a main steam line break, or a control element assembly ejection accident. 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, so that release of fission products to the

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BASES (continued)

APPLICAB.F SAFETY ANALYSES (continued)

environment is controlled by the rate of containment leakage. 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 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 10 CFR 100 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.

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

The single-failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with diverse power sources, motor-operated and pneumatically operated spring-closed, respectively. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.

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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	compromising containment OPERABILITY as long as the system is operated in accordance with the subject LCO. The mini-purge valves are capable of closing under accident conditions. Therefore, they are allowed to be open for limited periods during power operation. The containment isolation valves and 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 valve 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. 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 valves actuate on an automatic isolation signal. The valves covered by this LCO are listed with their associated stroke times in the FSAR (Ref. 3). The normally closed isolation valves or blind flanges are considered OPERABLE when they are locked-closed (manual valves), 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 or devices are those listed in Reference 4. 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 (continued)

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BASES (continued)

LCO (continued)

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 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 a containment isolation valve 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 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 3.9.3, "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 at the controls of the valve a dedicated operator, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when 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.

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BASES (continued)

APPLICABILITY (continued)

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

With one or more of the containment isolation valves 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 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.

In the event 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 the 4-hour Completion Time. 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.

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BASES (continued)

ACTIONS (continued)

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 required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. The Completion Time 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 every 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.

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 GDC 57). The Required Actions for Condition A assume two valves in a series are used to isolate the containment penetration and satisfy single-failure concerns.

Required Action A.1 has been modified by a Note stating that 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 A.2.1. These

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BASES (continued)

ACTIONS (continued)

systems are very small piping lines, such as instrument lines, which are a closed system 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

With one or more containment isolation valves 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 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 ensuring containment OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action B.2.1, the affected penetration must be verified to be isolated on a periodic basis. This is necessary to ensure containment OPERABILITY is maintained and that containment penetrations required to be isolated following an accident are isolated. The once per 31 days Completion Time for verifying that each affected penetration is isolated is appropriate considering 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 those 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 GDC 57 (Ref. 1). GDC 57 allows lines that enter containment and that are not part of the reactor coolant pressure boundary nor connected directly to containment atmosphere to be isolated by means of one containment isolation valve.

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ACTIONS (continued)

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

In the event 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 be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, 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 so 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 every 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 every 184 days, based on an NRC initiative, Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 5). Since more reliance is placed on a single valve while in this condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen.

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. This can be accomplished by entering the

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BASES (continued)

ACTIONS (continued)

supported systems' LCOs. [Alternatively, the appropriate Required Actions for the suppported 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 isolated in a penetration flow path associated with a supported ESF system, then the other penetration flow paths associated with the redundant counterpart supported ESF systems and their supported 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 actions 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

The plant must be placed in a MODE in which the LCO does not apply if the Required Actions and associated Completion Times are not met. 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.

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BASES (continued)

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 Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to prevent offsite dose limits from exceeding 10 CFR 100 limits (Ref. 2) or some fraction as established in the NRC staff-approved licensing basis following the most limiting DBA. Therefore, these valves are required to be in the sealed-closed position during MODES 1, 2, 3, and 4. A containment purge valve that is sealed-closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The Surveillance interval is a result of an NRC initiative, Generic Issue B-24 (Ref. 6), related to containment purge valve use during plant operations.

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), and air quality considerations for personnel entry, and for 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 interval is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that all containment isolation manual valves and blind flanges that are 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 the

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 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 administrative controls consist of stationing at the controls of the valve a dedicated operator, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when 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 requires verification that all containment isolation manual valves and blind flanges that are 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 the containment boundary is within design limits. For valves inside containment, the Surveillance interval specified as "prior to entering MODE 4 from MODE 5 if not performed more often than once per 92 days" is appropriate, since these

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

valves and flanges are operated under administrative controls and the probability of their misalignment is low.

A Note that allows normally locked- or sealed-closed isolation valves to be opened intermittently under administrative controls has been added to this SR. The administrative controls consist of stationing at the valve controls a dedicated operator, who is in continuous communication with the control room. 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 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 accordance with the Inservice Inspection and Testing Program, but 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 each automatic containment isolation valve will actuate to its isolation position on a containment isolation actuation 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 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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. 7), 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 seal types. Based on this observation, and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," (Ref. 5).

Additionally, this SR must be performed within 92 days after opening the valve. Ninety-two days 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.

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 excessive containment purge valve leakage is properly accounted for in determining the overall containment leakage rate to verify containment OPERABILITY.

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, "Compatibility for Containment Leakage Rate Testing;"

   General Design Criterion 53, "Provisions for Containment Inspection and Testing;"

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BASES (continued)

REFERENCES (continued)

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 [ ], "[Title]."
4. [Unit Name] FSAR, Section [ ], "[Accident Analysis]."
5. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."
6. Generic Issue B-24, "Containment Purge Valve Reliability."
7. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4A Containment Pressure (Atmospheric)

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 requirement 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 main steam line break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the containment Spray System.

Containment pressure is a process variable that is monitored and controlled during MODES 1, 2, 3, and 4. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a DBA, a loss of containment OPERABILITY may result. Loss of containment OPERABILITY could cause site-boundary doses, due to a DBA, to exceed values of the licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analyses and evaluations considered both LOCAs and MSLBs for determining the maximum peak containment pressure (P_a) of [55.7] psig. An MSLB at 102% THERMAL POWER results in the highest calculated internal containment pressure, [55.7] psig, which is below the internal design pressure of [60] psig. The MSLB event bounds the LOCA event from the containment peak pressure standpoint. It is this maximum containment pressure that is used to ensure that the licensing basis dose limitations are met.

The initial pressure condition used in the containment analysis was [14.7] psia ([0.0] psig). This resulted in a maximum peak pressure from a LOCA of [55.7] psig. The LCO

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

of [1.5] psig ensures that, in the event of an accident, the maximum accident design pressure for containment, [60] psig, is not exceeded. If an MSLB occurred while the containment internal pressure was at the LCO value of [1.5] psig, a total pressure of [57.3] psig would result. This value is still below the design value of [60] psig. The containment was also designed for an internal pressure equal to [5.0] psig below external pressure in order to withstand the resultant pressure drop from an accidental actuation of the Containment Spray System. The LCO limit of [-0.3] psig ensures that operation within the design limit of [-0.5] psig is maintained. The maximum calculated external pressure that would occur as a result of an inadvertent actuation of the containment Spray System is [2.8] psig.

Containment pressure satisfies Criterion 2 of the NRC Interim Policy Statement.

LCO

Maintaining containment pressure less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System. With containment pressure maintained within the limits of this LCO, containment OPERABILITY is ensured.

[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 the containment pressure channel inoperable and their justification are as follows:]

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BASES (continued)

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 MODES 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 within 1 hour. The Required Action must be taken in order to return operation to within the bounds of the containment analysis. The specified time period is consistent with the ACTIONS of LCO 3.5.1, "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 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. Based on operating experience, the allowed Completion Times are reasonable 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 the containment pressure is within limits ensures that facility operation remains within the limits assumed in the accident analysis. The 12-hour Frequency of

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

this SR was developed after taking into consideration operating experience related to both trending of containment pressure variations and pressure instrument drift during the applicable MODES. Furthermore, the 12-hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. 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 3.6.4B Containment Pressure (Dual)

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 main steam line break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential, with respect to the outside atmosphere, in the event of inadvertent actuation of the Containment Spray System. Containment pressure is a process variable that is monitored and controlled during MODES 1, 2, 3, and 4. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a DBA, 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 integrity are the LOCA and MSLB. The DBA LOCA and MSLB are analyzed using computer codes designed 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 Containment Spray System and one train of the Containment Cooling System being rendered inoperable).

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The containment analyses (Ref. 2) shows that the maximum peak calculated containment pressure, P_a, results from the limiting MSLB at 75% THERMAL POWER.

The initial pressure condition used in the containment analysis was [14.7] psig. The maximum containment pressure resulting from the limiting DBA, [42.3] psig, does not exceed the containment design pressure, [44] psig. The containment was also designed for an internal pressure equal to [0.65] psid below external pressure to withstand the resultant pressure drop from an accidental actuation of the Containment Spray System. The LCO limit of [27 inches of water] ensures that operation within the design limit of [-0.65] psid is maintained. The maximum calculated differential pressure that would occur as a result of an inadvertent actuation of the Containment Spray System is [0.49] psid.

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 less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure greater than or equal to the LCO lower pressure limit ensures the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.

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BASES (continued)

LCO (continued)	With containment pressure maintained within the limits of this LCO, containment OPERABILITY is ensured. [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 the containment pressure 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. 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 protect containment OPERABILITY.
ACTIONS	A.1 When containment pressure is not within the limits of the LCO, containment pressure 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. 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

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BASES (continued)

ACTIONS (continued)	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 facility operation remains within the limits assumed in the containment analysis. The 12-hour Frequency of this SR was developed after taking into consideration operating experience related to both trending of containment pressure variations and pressure instrument drift during the applicable MODES. Furthermore, the 12-hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.
REFERENCES	1. 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]." 3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 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 main steam line break (MSLB). In addition, though equipment installed inside containment is designed to operate at a higher temperature than allowed by this LCO, this temperature limit may help to minimize component degradation that may result from extended exposure to a high-temperature environment (Ref. 2).

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 on 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 (Ref. 3). 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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

environmental qualification operating envelope for both pressure and temperature.

The limit for containment average air temperature ensures that operation is maintained within the DBA analysis assumptions for containment. The accident analyses and evaluations considered both LOCAs and MSLBs for determining the maximum peak containment pressures and temperatures. The worst-case MSLB generates larger mass and energy releases than the worst-case LOCA. Thus, the MSLB event bounds the LOCA event from the containment peak pressure and temperature standpoint. The initial pre-accident temperature inside containment was assumed to be [120]°F (Ref. 2).

The initial containment average air temperature condition of [120]°F resulted in a maximum vapor temperature in containment of [413]°F. The temperature of the containment steel liner and concrete structure reach approximately 230°F and 220°F, respectively. The containment average air temperature limit of [120]°F ensures that, in the event of an accident, the maximum design temperature for containment, [300]°F, is not exceeded. The consequence of exceeding this design temperature may be the potential for degradation of the containment structure under accident loads.

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 maintained below the containment design temperature. As a result, the ability of containment to perform its 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:]

(continued)

BASES (continued)

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 of the LCO is not required in MODE 5 or 6 to ensure containment OPERABILITY.

ACTIONS

A.1

With containment average air temperature not within the limit of the LCO, containment average air temperature must be restored within 8 hours. The Required Action must be taken to return operation to within the bounds of the containment analysis. The 8-hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems 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 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 allowable Completion Times are reasonable, based on operating experience, to reach the required MODES from full power without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying the containment average air temperature is within the LCO limit ensures that containment operation remains within the limits assumed for the containment analysis. In

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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

REFERENCES

1. 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]."

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6A Containment Spray and Cooling Systems (Atmospheric & Dual)
(Credit taken for iodine removal by 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 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 and cooling systems are designed to the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," and GDC 40, "Testing of Containment Heat Removal Systems," 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, each capable of meeting the iodine-removal design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ENGINEERED SAFETY FEATURES (ESF) bus. The refueling water tank (RWT) supplies borated water to the containment spray during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWT to the containment sump(s).

The Containment Spray System provides a spray of cold borated water into the upper regions of the containment to reduce containment pressure and temperature during a DBA. The RWT 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 shutdown cooling heat

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BASES (continued)

BACKGROUND (continued)

exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet 50% of the system-design requirements for containment heat removal.

The Spray Additive System injects a hydrazine (N_2H_4) solution into the spray. The resulting alkaline pH of the spray enhances its ability to scavenge fission products from the containment atmosphere. The N_2H_4 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 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-High pressure signal coincident with a safety injection actuation signal (SIAS), or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two containment spray system pumps, and begins the injection phase. The Containment spray header isolation valves open upon a containment spray actuation signal. 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 RWT level Low signal is received. The Low level for the RWT generates a recirculation actuation signal (RAS) that aligns valves from the containment spray pump suction to the containment sump and/or signals the operator to manually align the system to the 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

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BASES (continued)

BACKGROUND (continued)

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.

Two trains of containment cooling each of sufficient capacity to supply 50% of the design cooling requirement, are provided. Two trains with two fan units each are supplied with cooling water from a separate train of service water cooling. All four fans are required to furnish the design cooling capacity. Air is drawn into the coolers through the fans and discharged to the steam generator compartments and pressurizer compartment.

In post-accident operation, following a containment cooling actuation signal (CCAS), all four Containment Cooling System fans are designed to start automatically in slow speed. Cooling is shifted from the chilled-water-cooled coils to the service-water-cooled coils. The temperature of the service water 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 OPERABILITY are the loss-of-coolant accident (LOCA) and the main steam line break (MSLB). The DBA LOCA and MSLB 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 with 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 and Containment Cooling System being inoperable.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The analysis and evaluation show that under the worst-case scenario, the highest peak containment pressure is [55.7] psig (experienced during an MSLB). The analysis shows that the peak containment vapor temperature is [413]°F (experienced during an MSLB). Both results are within the intent of the design basis. (See Bases B 3.6.4A and B 3.6.4B, "Containment Pressure," and B 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of [102]% RATED THERMAL POWER (RTP), one containment spray train and one containment cooling train operating, and initial (pre-accident) conditions of [120]°F and [14.7] psia. The analyses also assume a response-time delayed initiation in order to provide a conservative calculation of peak containment pressure and temperature responses.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a [-2.8] 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 B 3.6.4A and B 3.6.4B, "Containment Pressure." Inadvertent spray actuation in a dual containment will be addressed on a plant-specific basis.

The modeled Containment Spray System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint coincident with an SIAS to achieving full flow through the containment spray nozzles. The Containment Spray System total response time of [6] seconds includes diesel generator startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 2).

The performance of the containment cooling train for post-accident conditions is given in Reference 3. The result of the analysis is that each train can provide 50% of the required peak cooling capacity during the post-accident condition. The train post-accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 4.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The modeled Containment Cooling System actuation from the containment analysis is based upon the plant-specific response time associated with exceeding the CCAS to achieving full Containment Cooling System air and safety-grade cooling water flow.

The containment spray and cooling systems satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

During a DBA, a minimum of two containment cooling trains or two containment spray trains, or one of each, is required to maintain the containment peak pressure and temperature below the design limits (Ref. 5). 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 units must be OPERABLE. Therefore, in the event of an accident, minimum requirements are met, assuming that the worst-case single active failure occurs.

Each Containment Spray System typically includes spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT on an ESF actuation signal and automatically and/or manually 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 Spray System and an OPERABLE Containment Cooling System constitute the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure Containment Spray System and Containment Cooling System OPERABILITY:]

[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:]

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BASES (continued)

LCO (continued)

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. The probability and consequences of these events in MODES 5 and 6 are reduced due to the pressure and temperature limitations of these MODES. Thus, the containment spray and containment cooling systems 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 containment-cooling functions. The 72-hour Completion Time takes into account the redundant heat-removal capability afforded by the Containment Spray System, reasonable time for repairs, and low probability of a DBA occurring during this period.

B.1 and B.2

If the inoperable Containment Spray System 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 84 hours.

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BASES (continued)

ACTIONS (continued)

The 6 hours allotted to reach MODE 3 is a reasonable amount of 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 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 that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

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

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 are capable of providing greater than 100% of the heat-removal needs after an accident and provide iodine-removal capabilities. The 72-hour Completion Time was developed taking into account the redundant heat-removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the 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 combination of three or more Containment Spray System and Containment

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(continued)

BASES (continued)

ACTIONS (continued)

Cooling System 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

If the Required Actions and associated Completion Times for Condition C or D of this LCO 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 3.6.6A.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 position prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31-day Frequency of this SR was developed based upon Inservice Inspection and Testing Program requirements to perform valve testing at least once every 92 days. This SR does not require any testing or valve manipulation. Rather, it involves verifying, 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 and corrective action taken. The 31-day Frequency of this SR was developed considering the known reliability of the fan units and controls, the two-train

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(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

redundancy available, and the low probability of a significant degradation of the containment cooling train occurring between Surveillances, and has been shown to be acceptable through operating experience.

SR 3.6.6A.3

Verifying a service water flow rate of ≥ [2000] gpm to each cooling unit provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 2). The 31-day Frequency of this SR was based on Inservice Inspection and Testing Program requirements to perform testing on safety-related components at least once per 92 days. Also considered in selecting this Frequency were the known reliability of the cooling water system, the two-train redundancy, and the low probability of a significant degradation of flow occurring between Surveillances.

SR 3.6.6A.4

Verifying that the containment spray header piping is full of water to the [100-ft] level minimizes the time required to fill the header. This ensures that spray flow will be admitted to the containment atmosphere within the time frame assumed in the containment analysis. The 31-day Frequency is based on the static nature of the fill header and the low probability of a significant degradation of water level in the piping occurring between Surveillances.

SR 3.6.6A.5

Demonstrating that each containment spray pump develops ≥ [250] 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. 6). 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 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 3.6.6A.6 and SR 3.6.6A.7

These SRs demonstrate 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 signal. The containment spray valves also must actuate on an RAS. 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.8

This SR demonstrates that each containment cooling unit actuates on receipt of an actual or simulated actuation signal. The 18-month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6A.6 and SR 3.6.6A.7, above, for further discussion of the basis for the 18-month Frequency.

SR 3.6.6A.9

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. Performance of this SR demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive 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 spray nozzles.

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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]."

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

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

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

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 3.6.6B Containment Spray and Cooling Systems (Atmospheric & Dual) (Credit not taken for iodine removal by 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., specified fraction of 10 CFR 100 limits). The containment spray and cooling systems are designed to the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," and GDC 40, "Testing of Containment Heat Removal Systems," 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 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 tank (RWT) supplies borated water to the containment spray during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWT 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 RWT 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 shutdown cooling heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet 50% of the system design requirements for containment heat removal.

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BASES (continued)

BACKGROUND (continued)

The Containment Spray System is actuated either automatically by a containment High-High pressure signal coincident with a safety injection actuation signal (SIAS) or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two containment spray pumps, and begins the injection phase. The containment spray header isolation valves open on a containment spray actuation signal. 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 RWT level Low signal is received. The Low level signal for the RWT generates a recirculation actuation signal (RAS) that aligns valves from 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 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 50% of the design cooling requirements, are provided. Two trains with two fan units each are supplied with cooling water from a separate train of service water. All four fans are required to furnish the design cooling capacity. Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartments, [and outside the secondary shield in the lower areas of containment].

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BASES (continued)

BACKGROUND (continued)

In post-accident operation, following a containment cooling actuation signal (CCAS), all four Containment Cooling System fans are designed to start automatically in slow speed, if not already running. Cooling is shifted from the chilled-water-cooled coils to the service-water-cooled coils. The temperature of the service water 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 OPERABILITY are the loss-of-coolant accident (LOCA) and the main steam line break (MSLB). The DBA LOCA and MSLB 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 and Containment Cooling System being rendered inoperable.

The analysis and evaluation show that under the worst-case scenario, the highest peak containment pressure is [55.7] psig (experienced during an MSLB). The analysis shows that the peak containment vapor temperature is [414]°F (experienced during an MSLB). Both results are within the intent of the design basis. (See Bases B 3.6.4A and B 3.6.4B, "Containment Pressure," and B 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of [102]% RATED THERMAL POWER (RTP), one containment spray train and one containment cooling train operating, and initial (pre-accident) conditions of [120]°F and [14.7] psia. The analyses also

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

assume a response-time-delayed initiation in order to provide conservative peak calculated containment pressure and temperature responses.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a [-2.8] 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 B 3.6.4A and B 3.6.4B, "Containment Pressure." Inadvertent spray actuation in a dual containment will be addressed on a plant-specific basis.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint coincident with an SIAS to achieving full flow through the Containment spray nozzles. The Containment Spray System total response time of [6] seconds includes diesel generator startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 2).

Containment cooling train performance for post-accident conditions is given in Reference 3. The result of the analysis is that each train can provide 50% of the required peak cooling capacity during the post-accident condition. The train post-accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference [4].

The modeled Containment Cooling System actuation from the Containment analysis is based on the plant-specific response time associated with exceeding the CCAS to achieving full Containment Cooling System air and safety-grade cooling water flow.

The containment spray and cooling systems satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

During a DBA, a minimum of two containment cooling trains or two containment spray trains, or one of each, is required to maintain the containment peak pressure and temperature below

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BASES (continued)

LCO (continued)

the design limits (Ref. 5). To ensure that these requirements are met, two containment spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs.

Each Containment Spray System typically includes spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT upon an ESF actuation signal and automatically and/or manually 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 Spray System and an OPERABLE Containment Cooling System constitute the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure Containment Spray System and Containment Cooling System OPERABILITY:]

[For this facility, those required support systems which upon their failure do not require declaring the containment spray and cooling system 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 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

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BASES (continued)

APPLICABILITY (continued)

OPERABLE. Justification of Containment Spray System removal will be addressed on a plant-specific basis. The probability and consequences of these events in MODES 5 and 6 are reduced due to the pressure and temperature limitations of these MODES. Thus, the containment spray and containment cooling systems 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 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 developed taking into account the redundant heat-removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

B.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 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 developed based on the same reasons as those for Required Action A.1.

C.1

With two of the required containment spray trains inoperable, one of the required containment spray trains 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 developed taking into account the redundant heat-removal capabilities

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BASES (continued)

ACTIONS (continued)

afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

D.1 and D.2

With one of the required containment spray trains inoperable and one of the required containment cooling trains 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 greater than 100% of the heat-removal needs (for the condition of a one containment spray train inoperable and one containment cooling train inoperable) after an accident. The 72-hour Completion Time was developed based on the same reasons as those for Required Action C.1.

E.1

With two 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 are capable of providing greater than 100% of the heat-removal needs after an accident. The 72-hour Completion Time was developed based on the same reasons as those for Required Action C.1.

F.1

With any combination of three or more Containment Spray System and Containment Cooling System 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 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, 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 3.6.6B.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 also does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct positions prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. 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 (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 that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31-day Frequency was developed considering the known reliability of the fan units and controls, the two-train redundancy available, and the low probability of a significant degradation of the containment cooling train occurring between Surveillances.

SR 3.6.6B.3

Verifying a service water flow rate of ≥ [2000] gpm to each cooling unit provides assurance the design flow rate assumed in the safety analyses will be achieved (Ref. 2). The 31-day Frequency of this SR was based on Inservice Inspection and Testing Program requirements to perform testing on safety-related components at least once per 92 days. Also considered in selecting this Frequency were the known

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

reliability of the cooling water system, the two-train redundancy, and the low probability of a significant degradation of flow occurring between Surveillances.

SR 3.6.6B.4

Verifying the containment spray header is full of water to the [100-ft] level minimizes the time required to fill the header. This ensures that spray flow will be admitted to the containment atmosphere within the time frame assumed in the containment analysis. The 31-day Frequency is based on the static nature of the fill header and the low probability of a significant degradation of the water level in the piping occurring between Surveillances.

SR 3.6.6B.5

Demonstrating that each containment spray pump develops ≤ [250] 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. 6). 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 indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Inspection and Testing Program.

SR 3.6.6B.6 and SR 3.6.6B.7

These SRs demonstrate each automatic containment spray valve actuates to its correct position and each containment spray pump starts on receipt of an actual or simulated actuation signal. The containment spray valves also must actuate on an RAS. 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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.8

This SR demonstrates each containment cooling unit actuates on receipt of an actual or simulated actuation signal. The 18-month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6B.6 and SR 3.6.6B.7, above, for further discussion of the basis for the 18-month Frequency.

SR 3.6.6B.9

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. Performance of this SR demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive 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 spray nozzles.

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]."
3. [Unit Name] FSAR, Sections [ ], "[Title]."
4. [Unit Name] FSAR, Section [ ], "[Title]."
5. [Unit Name] FSAR, Section [ ], "[Title]."

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BASES (continued)

REFERENCES (continued)

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 3.6.7 Spray Additive System (Atmospheric & Dual)

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission-product inventory in the containment atmosphere in the event of an accident such as a loss-of-coolant accident (LOCA). 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 Part 100 (Ref. 1) or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

The addition of a spray additive to the boric acid spray solution increases the pH of the spray solution and maintains the containment sump pH above 8.0 during the recirculation phase of an accident. An elevated pH is desired since it enhances the iodine-removal capacity of the sprays and aids in the retention of iodine in the water in the Containment sump.

The Spray Additive System consists of a single spray chemical addition tank (SCAT) and two redundant 100%-capacity trains. Each train contains a chemical addition pump, an injection valve, isolation valves, a flow meter, and a flow controller. Upon receipt of a containment spray actuation signal (CSAS), the chemical addition pumps start and the injection valves open in each redundant train. The spray additive is then injected into the Containment Spray System at the suction of the containment spray pumps at metered amounts corresponding to the individual containment spray pump discharge flow rate. The rate at which the spray additive is added is reduced when a recirculation actuation signal is generated and the Containment Spray System enters the recirculation mode of operation. The pH of the containment spray solution is maintained between 9.0 and 10.0 during the injection mode and between 8.0 and 9.0 in the recirculation mode. Upon reaching a low-low level in the SCAT, the spray chemical addition pumps stop and the injection and isolation valves close (Ref. 2).

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BASES (continued)

BACKGROUND (continued)

The Spray Additive System reduces the iodine fission-production 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 effective removal of airborne iodine within containment following a Design Basis Accident (DBA).

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 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 spray additive tank volume is added to the remaining Containment Spray System flow path.

During a LOCA, the iodine inventory released to the containment is considered to be released instantaneously and uniformly distributed in the containment free volume. The containment volume is made up of sprayed and unsprayed regions. The sprayed region is enveloped by direct spray and mixed by the dome air circulators and emergency fan coolers. Mixing between the sprayed and unsprayed regions is facilitated by the emergency fan coolers and condensation of steam by the sprays.

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.

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BASES (continued)

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 maintain the pH of the spray solution between [9.0 and 10.0] in the injection mode and [8.0 and 9.0] in the recirculation mode. This pH range maximizes the effectiveness of the iodine-removal mechanism, without introducing conditions that may induce caustic stress corrosion cracking of mechanical components.

During a LOCA, one Spray Additive System train is capable of providing 100% of the required iodine-removal capacity. To ensure at least one train is available in the event of the limiting single failure, both trains must be maintained in an OPERABLE status.

[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:]

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.

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 the post-accident

(continued)

(continued)

BASES (continued)

APPLICABILITY (continued)

release of radioactive material 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 limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With the Spray Additive System inoperable, the system must be restored to OPERABLE status within 72 hours. The inoperability of the Spray Additive System includes the loss of capability to inject NAOH to either Containment Spray System suction line or both lines. The pH adjustment of the containment spray 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 paths capabilities, and the low probability of the worst-case 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 in the event the Spray Additive System is not restored to OPERABLE status within the associated Completion Time. 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 reasonable, 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 allows additional time for restoration of the Spray Additive System and is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the RCS.

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BASES (continued)

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 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 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the refueling water tank (RWT) 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 hydrazone (N_2H_4) solution in the Spray Additive System. The 184-day Frequency is based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal plant operations). Tank level is also indicated and alarmed in the control room, such that there is a high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the N_2H_4 concentration in the SCAT and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The concentration of N_2H_4 in the SCAT must be determined by chemical analysis. The 184-day Frequency is sufficient to ensure that the concentration level of N_2H_4 in the SCAT remains within the established limits. This is based on the

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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

The chemical addition pump must be demonstrated to provide the flow rate assumed in the accident analysis to the Containment Spray System. The Spray Additive System is not operated during normal plant operations. This prevents periodically subjecting systems, structures, and components within containment to a caustic spray solution. Therefore, this test must be performed on recirculation with the discharge flow path from each spray chemical addition pump aligned back to the SCAT. The differential pressure obtained by the pump on recirculation is analogous to the full spray add flow provided to the Containment Spray System on an actual CSAS. The Frequency of this SR is in accordance with the Inservice Inspection Testing Program and is sufficient to identify component degradation that may affect flow rate.

SR 3.6.7.5

This SR demonstrates that each automatic valve in the Spray Additive System flow path actuates to its correct position on a CSAS. 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 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 3.6.7.6

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 per 5 years. This SR provides assurance that the correct amount of N_2H_4 will be metered into the flow path upon Containment Spray System initiation. Due to

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	the passive nature of the spray additive flow controls, the 5-year Frequency is sufficient to identify component degradation that may affect flow rate.
REFERENCES	1. 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]."

DRAFT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Monitors—MODES 1 & 2 (Atmospheric & Dual)

BASES

BACKGROUND

Hydrogen monitors are required to monitor the hydrogen concentration in the containment following a loss-of-coolant accident (LOCA) or main steam line break (MSLB) in containment. Hydrogen may accumulate 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 an MSLB 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.

The hydrogen monitors are a post-accident Type A, Category 1, instrument. As such, they will function to allow monitoring of hydrogen following a LOCA or MSLB in containment.

Two independent hydrogen monitors have been provided and 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. When actuated the monitors will continuously monitor hydrogen concentration levels between 0 and 10%. Both monitors have the capability to interface with two areas that have been 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., to actuate the hydrogen recombiners or Hydrogen Purge System in accordance with emergency

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BASES (continued)

BACKGROUND (continued)

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 up 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. This information is used 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 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;

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 solution.

The hydrogen monitors satisfy Criterion 3 of the NRC Interim Policy Statement.

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BASES (continued)

LCO

Two hydrogen monitors must be OPERABLE with power from two independent safety-related power supplies. Either monitor 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 monitors 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 allow, if required, action to be taken to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA (Ref. 2). This ensures containment OPERABILITY and prevents damage to 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 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.

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BASES (continued)

APPLICABILITY (continued)

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 hydrogen monitors are not required in these MODES to ensure containment OPERABILITY.

ACTIONS

A.1

With one hydrogen monitor inoperable, the inoperable monitor 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 did 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 immediately entered.

B.1

The plant must be placed in a MODE in which the LCO does not apply if an inoperable hydrogen monitor 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 3.6.8.1

A CHANNEL FUNCTIONAL TEST is performed on each hydrogen monitor every 92 days in order to ensure that the entire

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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, a CHANNEL FUNCTIONAL TEST constitutes the following:]

SR 3.6.8.2

Performance of a CHANNEL CALIBRATION on the hydrogen monitors using sample gases ensures that the OPERABILITY of the monitors is maintained. A typical CHANNEL CALIBRATION includes a minimum of two data points to verify accuracy of the monitors 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. 4). Therefore, calibration with these sample gases helps ensure that accurate information regarding containment hydrogen concentrations up to and including the flammability limit is available to the operators following a LOCA. For this unit, 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 that 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, Section [ ], "[Title]."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Hydrogen Recombiners—MODES 1 & 2 (Atmospheric & 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 10 CFR 50, 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 main steam line break (MSLB). 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 recombiners are provided. Each consists of controls located in the control room, a power supply, and a recombiner located in containment. The recombiners have no moving parts. Recombination is accomplished by heating a hydrogen-air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the unit. Air flows through the unit at 100 cfm with natural circulation in the unit providing the motive force. 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 Design Basis Accident (DBA). This control would prevent a containment-wide hydrogen burn, thus ensuring containment OPERABILITY and minimizing challenges to the OPERABILITY of safety-

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

related equipment located in containment. The limiting DBA relative to hydrogen generation is a LOCA.

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

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

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

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

d. Corrosion of metals exposed to containment Spray System and Emergency Core Cooling 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 in Reference 3 are used to maximize the amount of hydrogen calculated.

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 4.1 v/o (Ref. 3). The Hydrogen Purge System is similarly designed such that it is redundant 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.

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BASES (continued)

LCO (continued)

[For this facility, an OPERABLE hydrogen recombiner constitutes the following:] In addition, for a hydrogen recombiner to be considered OPERABLE all the SRs must be met.

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 could lead to the generation of an amount of hydrogen (the flammability limit exceeded), sufficient 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 is to ensure their immediate availability after the safety injection and scram actuated on a LOCA or MSLB initiation. In the post-accident LOCA or MSLB environment, the one hydrogen recombiner is required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o, 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

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BASES (continued)

APPLICABILITY (continued)

requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

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

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 MSLB 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 did occur, 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

Performance of a system functional test for each hydrogen recombiner ensures that the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR requires verification that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum for approximately 2 minutes and verified to be ≥ 60 kW.

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 that there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, and 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;

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 3.6.9.3

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, GDC 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 & Dual)

BASES

BACKGROUND

The HMS supports 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 10 CFR 50, GDC 41, "Containment Atmosphere Cleanup" (Ref.2), 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 local pocket of hydrogen above the flammable concentration and giving 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 and is designed to withstand a loss-of-coolant accident (LOCA) without loss of function. The system has two independent trains, each of which consists of two dome air circulation fans, motors, and controls. Each train is sized for [37,000] cfm. The two trains are initiated automatically on a containment cooling actuation signal (CCAS) or can be manually started from the control room. Each train is powered from a separate emergency power supply. Since each train can provide 100% of the mixing requirements, the system will provide its design function with a limiting single active failure.

The HMS accelerates the air mixing process between the upper dome space of the containment atmosphere during LOCA operations. It also prevents any hot-spot air pockets during the containment cooling mode and avoids any hydrogen concentration in pocket areas.

Hydrogen mixing within the containment is accomplished by the Containment Spray System, the containment emergency fan coolers, and the containment internal structure design, which permits convective mixing and prevents entrapment. The HMS, operating in conjunction with the Containment Spray System and the emergency fan coolers prevents localized

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BASES (continued)

BACKGROUND (continued)	accumulations of hydrogen from exceeding the flammability limit 4.1 volume percent (v/o).

APPLICABLE SAFETY ANALYSES

The HMS ensures containment OPERABILITY by providing the capability of controlling the bulk hydrogen concentration in containment less than the lower flammable concentration of 4.1 v/o following a Design Basis Accident (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 LOCO 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 System 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 HMS is 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 (Ref. 4).

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

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BASES (continued)

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 CCAS.

[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 MSLB initiation. In the post-accident LOCA or MSLB 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.

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.

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BASES (continued)

APPLICABILITY (continued)

In MODES 5 and 6, the probability and consequences of a LOCA are low due to the pressure and temperature limitations of those MODES. Therefore, 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 MSLB 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 the hydrogen monitors.

Concurrent failure of two HMS subsystems within a 30-day period is considered a low-probability event. If such a double failure did 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 entered immediately.

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 allowable 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 3.6.10.1

Operating each HMS train for ≥ 15 minutes ensures that the train is OPERABLE and that all associated controls are

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 ≥ [37,000] cfm ensures that the 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 the HMS responds properly to CCAS. The 18-month Frequency was developed considering that 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.

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, GDC 41, "Containment Atmosphere Cleanup."

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BASES (continued)

REFERENCES (continued)

3. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission.

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.11 Iodine Cleanup System (ICS) (Atmospheric & Dual)

BASES

BACKGROUND

The ICS is provided per GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to reduce the concentration and quality of fission product released to the containment atmosphere following a 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., specified fraction of 100 CFR 100 limits).

The ICS consists of two 100%-capacity separate and redundant trains. Each train includes a heater, cooling coils, pre-filter, moisture separator, a high efficiency particulate air (HEPA) filter, an 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 relative humidity 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 credited in the analysis. The system initiates filtered recirculation of the containment atmosphere following receipt of a containment isolation actuation signal (CIAS). The system design is described in Reference 3.

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.

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

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BASES (continued)

BACKGROUND (continued)

and adsorbers. Both the moisture separator and heater are important to the effectiveness of the charcoal adsorbers.

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. Service water is required to supply cooling water to the cooling coils.

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, to exceed the calculated 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 main steam line break (MSLB), or a control element assembly (CEA) ejection accident. In the analysis for each of these accidents, it is assumed that adequate containment leak tightness is intact at event initiation and potential leakage to the environment is controlled by the rate of containment leakage. Additionally, it is assumed that the amount of radioactive iodine release 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:

a. A member of the general public who remains at the exclusion-area boundary for 2 hours following the onset of the postulated fission-product release; or

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BASES (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 Reference 1 are a whole-body dose of 25 rem or a 300-rem dose to the thyroid from iodine exposure, or both, or the NRC staff-approved licensing basis (e.g., 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 disabling the other train. 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 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 coil, moisture separator, ductwork, valves, and dampers are OPERABLE;

d. Containment isolation actuation instrumentation is OPERABLE;

e. Service water is available to the cooling coils; and

f. SRs are met.

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

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BASES (continued)

LCO (continued)

[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, MSLB, or a CEA ejection accident.

Because these accidents are considered credible accidents in MODES 1, 2, 3, and 4, containment OPERABILITY must be maintained to ensure 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; and

b. The fact that, even with no ICS train in operation, iodine would still be removed from the Containment atmosphere through adsorption by the containment Spray System.

Concurrent failure of two ICS trains within the 7-day period is considered a low-probability event. If such a double

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BASES (continued)

ACTIONS (continued)

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 immediately entered.

B.1 and B.2

The plant must be placed in a MODE in which the LCO does not apply if the ICS train 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.11.1

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, and the iodine-removal capability of the containment Spray System independent of the ICS.

SR 3.6.11.2

The Ventilation Filter Testing Program (VFTP) (Specification 5.8.4S) encompasses all the ICS filter tests consistent with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing of the performance of the HEPA filter, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 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 exists through or around the charcoal adsorber bed.

c. Verification of the methyl iodine 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 ICS train and the pressure drop across the combined prefilters, HEPA filters, and charcoal adsorber banks are within the required limits.

SR 3.6.11.3

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 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. Furthermore, the SR interval was developed considering that the system equipment OPERABILITY is demonstrated on a 31-day Frequency by SR 3.6.11.1.

SR 3.6.11.4

The filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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";

   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. 02), "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 3.6.12 Shield Building (Dual)

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 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. Following a LOCA, the Shield Buildi Exhaust Air Cleanup System (SBEACS) 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. A description of the SBEACS is provided in Basis B 3.6.17. Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the SBEACS.
APPLICABLE SAFETY ANALYSES	The design basis for shield building OPERABILITY is a large-break 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 SBEACS, 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., specified fraction of 10 CFR 100 limits) during an accident.
LCO	Shield building OPERABILITY must be maintained to ensure proper operation of the SBEACS 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 main steam line break, LOCA, or control element assembly ejection accident could release radioactive material to the primary containment atmosphere.

In MODES 5 and 6 the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, neither containment OPERABILITY or shield building OPERABILITY is 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

The plant must be placed in a MODE in which the requirement does not apply if shield building OPERABILITY cannot be restored in the required time period. 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS	<u>SR 3.6.12.1</u> 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. <u>SR 3.6.12.2</u> 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 Surveillance Frequency of 31 days is based on engineering judgment and is considered adequate in view of other indications of door status available to the operator. <u>SR 3.6.12.3</u> This Surveillance 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 3.6 CONTAINMENT SYSTEMS

B 3.6.13 Vacuum Relief Valves (Dual)

BASES

BACKGROUND

The vacuum relief valves protect the containment vessel against negative pressure, i.e., a lower pressure inside than outside. Excessive negative pressure inside containment can occur if there is an inadvertent actuation of the Containment Cooling System or the Containment Spray System. Multiple equipment failures and/or human errors are necessary to have inadvertent actuation.

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., specified fraction of 10 CFR 100 limits).

The containment pressure vessel contains two vacuum breakers that protect the containment from excessive external loading. The vacuum breakers are 24-inch penetrations that connect the shield building annulus to the containment. Each penetration consists of a pneumatically operated butterfly valve in series with a check valve located on the containment side of the penetration.

Each butterfly valve is actuated by a separate pressure controller that senses the differential pressure between the containment and the annulus. Each butterfly valve is provided with an air accumulator that allows the valve to open following a loss of instrument air.

The combined pressure drop at rated flow through either vacuum breaker will not exceed the containment pressure vessel design external pressure differential of [0.65] psig with any prevailing atmospheric pressure.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Design of the vacuum relief valves involves calculating the effect of an inadvertent containment spray actuation that can reduce the atmospheric temperature (and hence pressure) inside containment (Ref. 2). Conservative assumptions are used for all the pertinent parameters in the calculation. For example, the minimum spray water temperature is assumed, as well as maximum initial containment temperature, maximum spray flow, all trains of spray operating, etc. The resulting containment pressure versus time is calculated, including the effect of the vacuum relief valves opening 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 [0.65] 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 [-0.368] psig. This resulted in a minimum pressure inside containment of [0.49] psig, which is less than the design load.

The vacuum relief valves must also perform the containment isolation function in containment high-pressure event. For this reason, the system is designed to take the full containment positive design pressure and the containment DBA 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 the Containment Spray System. Two vacuum relief valves are required OPERABLE to ensure that at least one is available, assuming the other valve fails to open. The pressure controller that senses differential pressure between the containment and the annulus must be OPERABLE, as it automatically actuates the butterfly valves. Instrument air is also required for butterfly valve actuation.

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BASES (continued)

LCO (continued)

[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 valve 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 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 or Containment Cooling System.

In MODES 5 and 6, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations of these MODES. The Containment Spray 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

With one of the required vacuum relief valves 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.

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BASES (continued)

ACTIONS (continued)

Concurrent failure of two vacuum relief valves would result in the loss of functional capability. Therefore, LCO 3.0.3 must be immediately entered.

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 references the Inservice Inspection Test Program, which establishes the requirement that inservice inspection of the American Society of Mechanical Engineers (ASME) Code 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 [ ], "[Title]."
3. 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 3.6.14 Shield Building Exhaust Air Cleanup System (SBEACS) (Dual)

BASES

BACKGROUND

The SBEACS is required by 10 CFR 50 Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials leaking from the primary containment of a dual 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., specified fraction of 10 CFR 100 limits).

The dual containment has a secondary containment, 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.

Following a LOCA, the SBEACS 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 SBEACS.

The SBEACS consists of two separate and redundant trains. Each train includes a heater, cooling coils, prefilter, moisture separator, a high-efficiency particulate air (HEPA) filter, an 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 of the main HEPA filter bank. Only the upstream HEPA filter and the charcoal

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BASES (continued)

BACKGROUND (continued)

adsorber section are credited in the analysis. The system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a safety injection actuation signal (SIAS). 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 operating in high humidity. Continuous operation of each train for at least 10 hours per month with heaters on reduces moisture buildup on the HEPA filters and adsorbers. The cooling coils cool the air to keep the charcoal beds from becoming too hot due to absorption of fission products.

During normal operation the Shield Building Cooling System is aligned to bypass the SBEACS's HEPA filters and charcoal adsorbers. For SBEACS operation following a DBA, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers.

The SBEACS reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the SBEACS 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 SBEACS 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 SBEACS 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.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The modeled SBEACS actuation in the safety analysis is based on a worst-case response time associated with exceeding an SIAS. 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 SBEACS satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

In the event of a DBA, one SBEACS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two trains of the SBEACS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

A train is 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 separator, ductwork, valves, and dampers are OPERABLE; and

d. SRs are met.

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

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

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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 SBEACS 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 SBEACS train and the low probability of a DBA occurring during the period.

Concurrent failure of two SBEACS trains would result in the loss of functional capability. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the SBEACS 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.14.1

Operating each SBEACS 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, and the iodine-removal capability of the Containment Spray System.

SR 3.6.14.2

The Ventilation Filter Testing Program (VFTP) (Specification 5.8.4S) encompasses all the SBEACS filter tests consistent with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing of the performance of the HEPA filter, 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. The following tests are included:

a. Verification of the in-place (cold) penetration and bypass dioctyl phthalate (DOP) test leakage of each SBEACS train. The cold DOP test co firms the validity of the pre-installation hot DOP tests 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 SBEACS train. This test determines that no bypass paths exist through or around the charcoal adsorber bed.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

c. Verification of the methyl-iodine 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 SBEACS train and the pressure drop across the combined prefilters, HEPA filters, and charcoal adsorber banks are within the required limits.

e. For systems with heaters, verification of the proper functioning of each SBEACS train's heaters.

SR 3.6.14.3

The automatic startup ensures that each SBEACS train responds properly. The 18-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 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 3.6.14.1.

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 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 ≥ [0.25]-inch water gauge during test operation within 1 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 18-month Frequency is consistent with the Regulatory Guide 1.52 (Ref. 5) guidance for functional testing.

SR 3.6.14.6

[For this facility, the purpose of this SR and justification of the Frequency interval are as follows:]

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."

2. Title 10, Code of Federal Regulations, Part 100, "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, Revision 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 3.7 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 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.

Eight MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in Reference 1. The MSSVs' rated capacity passes the full steam flow at 102% RATED THERMAL POWER (RTP) (100% + 2% for instrument error) with the valves full open. 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 number of valves needed will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open all valves following a turbine-reactor trip.

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 [16] 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 (RTP plus [14] Mwt pump heat input) is [17.83 E6] lb/hour. The ratio of this total capacity to the total steam flow is [109.2]%.

The low-pressure setpoint MSSV, [1000] psia, corresponds to a zero-power loop average temperature (T_{avg}) (secondary fluid saturation temperature) of [545]°F. The RCS T_{avg} must be above this temperature to open the MSSVs.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from the ASME Code; its purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated 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, and are presented in Section [15.2] of the FSAR (Ref. 4). Of these, the full-power loss-of-condenser vacuum (LOCV) event is the limiting AOO. An LOCV isolates the turbine and condenser, and terminates normal feedwater flow to the steam generators. Before delivery of auxiliary feedwater (AFW) to the steam generators, RCS pressure reaches ≤ 2,630 psig. This peak pressure is < 110% of the design pressure of 2,500 psig, but high enough to actuate the pressurizer safety valves. The maximum relieving rate during the LOCV event is 2.5 E6 lb/hour, which is less than the rated capacity of two of the MSSVs.

The limiting accident for peak RCS pressure is the full-power feedwater line break, inside containment, with the failure of the backflow check valve in the feedwater line from the affected steam generator. Water from the affected steam generator is assumed to be lost through the break with minimal additional heat transfer from the RCS. With heat removal limited to the unaffected steam generator, the reduced heat transfer causes an increase in RCS temperature, and the resulting RCS fluid expansion causes an increase in pressure. The RCS pressure increases to ≤ 2,730 psig, with the pressurizer safety valves providing relief capacity. The maximum relieving rate of the MSSVs during the feedwater line break event is ≤ 2.5 E6 lb/hour, which is less than the rated capacity of two of the MSSVs.

Using conservative analysis assumptions, a small range of feedwater line break sizes less than a full double-ended guillotine break produce an RCS pressure of 2,765 psig for a period of [20] seconds; exceeding 110% (2,750 psig) of design pressure. This is considered acceptable as RCS pressure is still well below 120% of design pressure where deformation may occur. The probability of this event is in the range of [4 E-6/year].

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

Two MSSVs per steam generator are required by the accident analysis to provide overpressure protection for design basis transients occurring at 102% of RTP. A MSSV will be considered inoperable, however, if it fails to open upon demand. The LCO requires all MSSVs to 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 full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements), and adjustment to the Reactor Protective System trip setpoints. These limitations are addressed in Table 3.7.1-1, Required Action A.2.2.1, and Required Action 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 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 function to mitigate the consequences of accidents that could result in a challenge to the reactor coolant pressure boundary.

APPLICABILITY

In MODE 1 above [ ]% RTP, the number of MSSVs per steam generator to be required OPERABLE must be as specified in Table 3.7.1-1. Below [ ]% RTP in MODES 1, 2, and 3, only two MSSVs are required to be OPERABLE per steam generator.

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

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BASES (continued)

APPLICABILITY (continued)

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES. Should the steam generators be water-solid, however, it is prudent to have overpressure protection for them.

ACTIONS

A.1

With less than the required number of MSSVs per Table 3.7.1-1, verify that at least two required MSSVs per steam generator are OPERABLE. Only two MSSVs are required in MODES 2 and 3, and only two in MODE 1 below [ ]% RTP. Above [ ]% 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 if 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 the information sources, such as maintenance logs, to determine if 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 as per Table 3.7.1-1. Based on operating experience, the 4-hour Completion Time to restore an MSSV to OPERABLE status is reasonable, and takes into account the relative importance of maintaining the OPERABILITY of these valves, and the low

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(continued)

BASES (continued)

ACTIONS (continued)

probability of an event occurring during this period that requires 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 power level. Operation may continue, provided the allowable THERMAL POWER and the [variable high power trip (VHPT)—high] setpoints are reduced by these formulas:

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

The ceiling on the variable overpower trip is also reduced to an amount over the allowable THERMAL POWER equal to the Band given for this trip in Table 3.3.1-1.

SP = RP x N + [9.8]

where:

RP = Reduced reactor THERMAL POWER as a percentage of RTP. This is a ratio of the available relieving capacity over the total steam flow at rated power;

SP = Reduced reactor trip setpoints calculated for LCO 3.7.1;

Y = Total OPERABLE MSSVs relieving capacity per steam generator based on a summation of the individual MSSV relief capacity in lb/hour;

N = ALLOWABLE VALUE for VHPT—High setpoints as specified in LCO 3.3.1, "Reactor Protection System;"

Z = Steam generator calculated steam flow rate at 100% RTP + 2% instrument uncertainty expressed in lb/hour (see text above); and

[9.8] = Band between the maximum THERMAL POWER and the variable overpower trip setpoint ceiling (Table 3.3.1-1).

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(continued)

BASES (continued)

ACTIONS (continued)

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

The 4-hour Completion Time for Required Action A.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 a period of 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 the occurrence of a transient that could result in steam generator overpressure during this period.

B.1 and B.2

If the MSSVs can not be restored to OPERABLE status in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit 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 fullpower operation in an orderly manner and without challenging the 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

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(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves tested every 24 months. The SRs are specified in the Inservice Inspection and Testing Program that encompasses Section XI of the ASME Code. This Code provides 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 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 [5.2], "[Overpressure Protection]."

2. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Article NC-7000, "Overpressure Protection," Class 2 Components.

3. [Unit Name] FSAR, Section [15.2], "[Accident Analysis—Decreased Heat Removal Events]."

4. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Article IWV-3500, "Inservice Tests—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)

Power Range Neutron Flux—High Trip Setpoint
versus OPERABLE MSSVs

MINIMUM NUMBER OF MSSVs PER SG REQUIRED OPERABLE	APPLICABLE POWER, % RTP	APPLICABLE TRIP SETPOINT, % RTP
5	> [87]	< [111.1]
4	< [87], ≥ [65]	< [87]
3	< [65], ≥ [43]	< [65]
2	< [43]	< [43]

Table 3.7.1-1 (Page 1 of 1)

MSSV Lift Setting

VALVE NUMBER				LIFT SETTING, PSIG, + 3%
SG #1	SG #2	[SG #3]	[SG #4]	

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high-energy line break. MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves, [automatic depressurization valves], and auxiliary feedwater pump turbine's steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, steam bypass system, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal (MSIS) generated by either low steam-generator pressure or high Containment pressure. The MSIVs fail closed on loss of control or actuation power. The MSIS also actuates the main feedwater isolation valves to close. 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 inside containment (Ref. 2). It is also influenced by the accident analysis of the steam line break 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 hot zero-power steam-line break inside containment with a loss-of-offsite power following turbine trip, and failure of the

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

MSIV or the affected steam generator to close. At zero power, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow, failure of the MSIV to close contributes to the total release of the additional mass and energy in the steam headers, which are downstream of the other MSIV. 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 borated water injection delivered by the Emergency Core Cooling System. Other failures considered are the failure of a MFIV to close, and failure of an emergency diesel generator to start.

The accident analysis compares several different steam line break events against different acceptance criteria. The large steam line break 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 steam line break 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 (RCS) cooldown. With a loss-of-offsite power, the response of mitigating systems, such as the high pressure safety injection pumps, is delayed. Significant single failures considered include: failure of a MSIV to close, failure of an emergency diesel generator, and failure of an HPSI pump.

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 the containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

accident scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator, and from the residual steam in the main steam header downstream of the closed MSIV in the intact loop.

b. A break outside of containment and upstream from the MSIVs. This scenario is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break, and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs. This type of break will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the steam bypass valves will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIVs isolates the affected steam generator from the intact steam generator. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the steam generator with the ruptured tube below the MSSV's 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. These events are 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.

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BASES (continued)

LCO (continued)

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 following support systems are required to be OPERABLE to ensure MSIV OPERABILITY:]

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

APPLICABILITY

The MSIVs must be OPERABLE whenever there is significant mass and energy in the RCS and steam generators. In MODES 1, 2, and 3, there is significant mass and energy in the RCS and steam generators; therefore, the MSIVs must be OPERABLE or closed. When the MSIVs are closed, they are already performing their safety function.

In MODE 4, the MSIVs are normally shut, and the steam generator energy is low.

In MODES 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water, therefore, the MSIVs are not required for isolation of potential high-energy secondary-system pipe breaks in these MODES.

A Note has been added to provide clarification that the Completion Time is on a Condition Basis; Condition 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 can be made to the MSIV with the plant hot. The 8-hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs.

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BASES (continued)

ACTIONS (continued)

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 immediately entered.

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 MODES 2 or 3 in one or more flow paths, restore MSIVs to OPERABLE status, or close inoperable MSIVs 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 are closed, the inoperable MSIVs must be verified to be continually closed on a periodic basis. This is necessary to ensure that the assumptions in the safety analysis 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 that will ensure that these valves will continue to be in the closed position.

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BASES (continued)

ACTIONS (continued)

D.1 and D.2

With two MSIVs inoperable in MODE 2 or 3, in the same flow path for one or more paths, restore at least one MSIVs 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 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 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 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 SR 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 with the plant 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 MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Inspection and Testing Program or 18 months,

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

whichever is less. This 18-month surveillance Frequency demonstrates the valve closure time at least once per refueling cycle. 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 test is conducted in MODE 3, with the plant at operating temperature and pressure, as part of the ASME 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 not applicable to this SR for entry into MODE 3, provided testing is complete within 24 hours after reaching acceptable test conditions. This allows delaying testing to MODE 3, in order to have 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. 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 3.7.3 Main Feedwater Isolation Valves (MFIVs) 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. Closure of the MFIVs and associated bypass valves terminates flow to both steam generators, terminating the event for feedwater-line breaks occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs and associated bypass valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or feedwater line breaks inside containment, and reducing the cooldown effects for SLBs.

The MFIVs and associated bypass valves isolate the non-safety-related portions from the safety-related portion 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 provides a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loop.

One MFIV is located on each AFW line, outside, but close to, containment. The MFIVs are located upstream of the AFW injection point so that AFW may be supplied to a steam generator following MFIV closure. The piping volume from the valve to the steam generator 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.

The MFIVs and its associated bypass valves close on receipt of a main steam isolation signal (MSIS) generated by either low steam generator pressure, or high containment pressure. The MSIS also actuates the main steam isolation valves (MSIVs) to close. The MFIVs and its associated bypass valves may also be actuated manually. In addition to the MFIVs and its associated bypass valves, a check valve inside containment is available to isolate the feedwater line

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BASES (continued)

BACKGROUND (continued)	penetrating containment, and to ensure that the consequences of events do not exceed the capacity of the containment heat removal systems. A description of the MFIVs is found in Reference 1.
APPLICABLE SAFETY ANALYSES	The design basis of the MFIVs is established by the analysis for the large SLB. It is also influenced by the accident analysis for the large feedwater-line break. Closure of the MFIVs and its associated bypass valves may also be relied on to terminate a steam break for core response analysis and an excess feedwater event upon receipt of a MSIS on high steam-generator level. Failure of an MFIV and its associated bypass valves to close following an SLB, feedwater-line break, or excess-feedwater event can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or feedwater-line break event. The MFIVs satisfy Criterion 3 of the NRC Interim Policy Statement.
LCO	Following a feedwater-line break or SLB, this LCO ensures that the MFIVs and their associated bypass valves will isolate MFW flow to the steam generators. These valves will also isolate the non-safety-related portions from the safety-related portions of the system. This LCO requires that the MFIV and their associated bypass valves in each of the feedwater lines be OPERABLE. The MFIVs 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 feedwater-line break inside containment. If an MSIS on high steam generator level is relied on to terminate an excess-feedwater-flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

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BASES (continued)

LCO (continued)

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

[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 their associated bypass valves 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 MODES 1, 2, and 3, the MFIVs 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 the case of a secondary system pipe break inside 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 their associated bypass valves are normally closed since MFW is not required. In MODES 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water. Therefore, the MFIVs and their associated bypass valves are not required for isolation of the potential high-energy secondary-system pipe breaks in these MODES.

For this LCO, Note has been to provide clarification that Conditions A and B are treated as an entity with a single Completion Time.

ACTIONS

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

With one MFIV or its associated bypass valve in one or more flow paths inoperable, restore affected valves to OPERABLE status, or close or isolate inoperable affected valves within [8 or 72] hours. When these valves are closed or

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BASES (continued)

ACTIONS (continued)

isolated, they are performing their required safety function (e.g., to isolate the line).

The typical arrangement assumed in these actions is a bypass valve in parallel with an MFIV. If an MFIV or bypass valve is inoperable and open under these conditions, then it is assumed that the closed system inside containment will function to isolate the line.

[(For plants with only one MFIV per feedwater line): The 8-hour Completion Time is a reasonable amount of time to complete the actions required to close the MFIV or its associated bypass valve, 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 MFIVs closed in an orderly manner and without challenging plant systems.]

The 72-hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, and the low probability of an event that would require isolation of the main feedwater flow paths occurring during this period.

For inoperable MFIVs and their associated bypass valves that cannot be restored to OPERABLE status within the specified Completion Time, but were closed or isolated, the inoperable valves must be verified that they continue to be closed or isolated on a periodic basis. This is necessary to ensure that the assumptions in the safety analysis 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.

B.1 and B.2

If more than one MFIV or its associated bypass valves 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

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BASES (continued)

ACTIONS (continued)

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 condition where 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 MFIVs or its associated bypass valve, 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 MFIVs closed in an orderly manner and without challenging plant systems.

C.1 and C.2

If the MFIVs and their associated bypass valves cannot be restored to OPERABLE status, closed, or isolated 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 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 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 MFIVs should not be tested at 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 is in accordance with the Inservice Inspection and Testing Program or 18 months,

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

whichever is less. The surveillance interval of 18 months for Operating experience has shown that these components usually pass the SR when performed on the 18-month Frequency.

SR 3.7.3.1 is modified by a Note that allows exemption to SR 3.0.4. SR 3.0.4 is not applicable to this SR for entry into MODE 3 provided testing is completed within 24 hours after reaching acceptable test conditions. This allows delaying testing to MODE 3 in order to establish conditions consistent with those under which the acceptance criteria were 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 (ADVs) (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; the turbine-driven pump provides [100]% 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 one steam generator, although each pump has the capability to be realigned from the control room to feed the other steam generator. The steam turbine-driven AFW pump receives steam from either main steam header upstream of the main steam isolation valve (MSIV). Each of the steam feed lines will supply 100% of the requirements of the turbine-driven AFW pump.

The turbine-driven AFW pump supplies a common header capable of feeding both steam generators, with dc-powered control valves actuated to the appropriate steam generator by the Emergency Feedwater Actuation System (EFAS). One pump at full flow is sufficient to remove decay heat and cool the plant to Shutdown Cooling System (SDC) entry conditions. Thus the requirement for diversity in motive power sources for the AFW system is met.

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BASES (continued)

BACKGROUND (continued)

The AFW System supplies feedwater to the steam generators during normal plant startup, shutdown, and hot standby conditions.

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 SDC entry conditions, and steam is released through the ADVs.

The AFW System actuates automatically on low steam generator level by the EFAS as described in LCO 3.3.2. The EFAS logic is designed to feed either or both steam generators with low levels, but will isolate the AFW system from a steam generator having a significantly lower steam pressure than the other steam generator. The EFAS automatically actuates the AFW turbine-driven pump and associated dc-operated valves and controls when required, to ensure an adequate feedwater supply to the steam generators. DC-operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System is discussed in Reference 1.

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a 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 make-up water to replace steam generator secondary inventory being lost as steam 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.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are:

a. Feedwater System pipe rupture; and

b. Loss of normal feedwater.

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of small-break loss-of-coolant-accident.

The AFW System design is such that it can perform its function following a feedwater line break between the main feedwater isolation valve 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 EFAS logic might 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 was terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The AFW System satisfies 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 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 diverse means, a steam-driven turbine supplied with steam from a source that is not isolated by the closure of the MSIVs.

The AFW System is considered OPERABLE when the components and flow paths required to provide AFW flow to the steam generators are OPERABLE. This requires that the two motor-driven AFW pumps be OPERABLE in [two] diverse paths, each

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BASES (continued)

LCO (continued)

supplying AFW to a separate steam generator. The turbine-driven AFW pump shall be OPERABLE with redundant steam supplies from each of the [two] main steam lines upstream of the MSIVs and capable of supplying AFW flow to either of the two Steam Generators. The piping, valves, instrumentation, and controls in the required flow paths shall be OPERABLE.

The LCO is modified by a Note requiring that only one motor-driven AFW pump be OPERABLE in MODE 4. This is because of reduced heat removal requirements, the short period of time in MODE 4 during which AFW is required, and the insufficient steam supply available in MODE 4 to power the turbine-driven AFW pump.

[For this facility, each OPERABLE AFW train consists of the following:]

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

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

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE and to function in the event that the main feedwater is lost. In addition, the AFW system is required to supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

In MODE 4, without any SDC loops OPERABLE and in operation, the AFW system is required to be used for heat removal via the steam generator.

In MODES 5 and 6, the steam generators are not used for decay-heat removal, and the AFW System is not required.

For this LCO, Note has been added to provide clarification that all components of the AFW trains are treated as an entity with a single Completion Time.

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BASES (continued)

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 requiring the inoperable steam supply to the turbine-driven AFW pump.

B.1

With one of the required AFW trains (pump or flow path) 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 in light of the redundant capabilities afforded by the AFW system, reasonable time for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and flow paths remain to supply feedwater to the steam generators.

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. Required Action C.2 has been modified by a Note intended to restrict entry into MODE 4 without any SDC loop 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 SDC loops become inoperable or are not in operation. The allowed Completion

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BASES (continued)

ACTIONS (continued)

Times are reasonable, based on operating experience, to reach the required MODES 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 AFW train is required in accordance with the Note that modifies the LCO. Although it is not required, the plant may continue to cooldown, and start the SDC, if prudent.

D.1

With all three AFW trains 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-grade 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. LCO 3.0.3 is not applicable as 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 water and steam flow paths, provides assurance that the proper flow paths exist for AFW 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 manipulations. Rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

The 31-day Frequency of this SR 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 to provide added assurance of valve correct 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. The ASME Section XI (Ref. 2) inservice testing (required only at 3-month intervals) satisfies this requirement when performed, as per Specification 6.8.1.j of the Inservice Inspection and Testing Program.

A 31-day Frequency on a Staggered Test Basis results in testing each pump once every 3 months as required by the ASME code.

SR 3.7.4.2 is modified by a Note that allows an exception to SR 3.0.4. The provisions of SR 3.0.4 are not applicable for entry into MODE 3 for purposes of testing the turbine-driven AFW pump because there is insufficient steam in MODES 4, 5, and 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 Emergency Feedwater Activation System signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. The actuation logic is tested every 92 days as part of the Safety Injection Actuation System (SIAS) functional test, except for the subgroup relays that actuate, the system cannot be tested during normal plant operations. The 18-month Frequency was developed considering it is prudent that these surveillances

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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, considering the design reliability of (and confirming operating experience using) the equipment.

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 EFAS signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The actuation logic is tested as part of the SIAS functional test every 92 days, except for the subgroup relays that actuate the system that 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 to suspend the provisions of SR 3.0.4 for entry into MODE 3 for purposes of testing the turbine-driven AFW pump because there is insufficient steam in MODES 4, 5, and 6 to perform a valid test. SR 3.0.4 is not applicable to this SR 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 System is properly aligned by demonstrating the flow path to each steam generator prior to entering MODE 2 operation, after 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 of other administrative controls 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. This SR ensures that the flow path from the CST to the steam generators is properly aligned by requiring a verification of minimum flow capacity of [750] gpm at 1270 psi. (This SR is not required by those plants that use AFW for normal startup and shutdown.)

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BASES (continued)

REFERENCES

1. [Unit Name] FSAR, Section [10.4.9], "[Auxiliary Feedwater System]."

2. American Society for Mechanical Engineers (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 3.7.5 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. The CST is designed to Seismic Category I requirements to ensure availability of the feedwater supply. Feedwater is also available from an alternate source. [For this facility, the alternate source(s) of feedwater and their safety-grade classification 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 (Ref. 2 and Ref. 3). For anticipated operational occurrences and accidents which 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 shutdown-cooling 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 with a coincident loss-of-offsite power. Single failures that also affect this event include:

a. The 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. The 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 non-limiting event considered in CST inventory determinations is a break either in the main feedwater, or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, as the Emergency Feedwater Actuation System (EFAS) would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated by the retaining 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 cooldown the RCS to shutdown cooling 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 the cooldown, as well as to account for any losses from the steam-driven AFW pump's turbine, or before isolating AFW to a broken line.

The level required is equivalent to a usable volume of [250,000] gallons, which is based on holding the plant in

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BASES (continued)

LCO (continued)

MODE 3 for 2 hours followed by a cool down to SCS entry conditions at 75°F/hour. This basis is established by Branch Technical Position, Reactor Systems Branch 5-1 (Ref. 4) and exceeds the volume required by the accident analysis.

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 the CST inoperable and their justification are as follows:]

APPLICABILITY

In MODES 1, 2, 3, and 4, the applicability of the CST is consistent with AFW System applicability, since the CST directly supports the AFW System.

In MODES 5 and 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1

With the CST unable to supply the required volume of cooling water to the AFW pumps, it must be restored to OPERABLE status. Four hours allows 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.

If equipment used to verify CST level is determined to be inoperable, the CST level is considered to be out of limits and Required Actions A.1 and A.2.2 Completion Times apply to restore such equipment to OPERABLE status.

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 expire.

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ACTIONS (continued)

OPERABILITY of the backup feedwater supply must include include verification of the OPERABILITY of flow paths from the backup supply to the AFW pumps, and availability of the required volume of water in the backup supply. The CST must still be returned to OPERABLE status within 7 days as 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 water supply. The 7-day Completion Time is reasonable in view of the fact that an OPERABLE backup water supply is available, and the low probability of an event requiring the use of the water from the CST occurring during this period. [For this facility, an OPERABLE backup water supply to the AFW System constitutes the following:]

B.1 and B.2

If the CST cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the requirements do not apply with the Shutdown Cooling System (SCS) in operation. 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 D.2) within 18 hours. This allows an additional 6 hours for the SCS to be placed in service after entering MODE 4. Required Action D.2 has been modified by a Note intended to restrict entry into MODE 4 without an RHR train 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 D.1, all RHR trains become inoperable or are 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

This SR verifies that the CST contains the required volume of cooling water. (This level may be a single value or a function of RCS conditions.) The 12-hour Frequency of this

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR was developed based on operating experience, and in view of 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 CST level deviations.

REFERENCES

1. [Unit Name] FSAR, Section [9.2.6], "[Condensate Storage and Transfer System]."

2. [Unit Name] FSAR, Section [6], "[Engineered Safety Features]."

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

4. NRC Standard Review Plan Branch Technical Position (BTP) 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 out-leakage from the Reactor Coolant System (RCS). Under steady-state conditions, the activity is primarily iodines with relatively short half-lives, and thus is indication of current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission-product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environs because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity valve 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 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 level, the resultant 2-hour thyroid dose to a person at the exclusion area boundary (EAB) would be about [1.3] rem should the main steam safety valves (MSSVs) open for the 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 limit 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 generator is available for core decay-heat dissipation by venting steam to the atmosphere through MSSVs and steam generator atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generator. Venting continues until the reactor coolant temperature and pressure has decreased sufficiently for the Shutdown Cooling System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through 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 3 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 100.11 (Ref. 1).

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BASES (continued)

LCO (continued)

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply whenever using the steam generators for RCS heat removal. This is a potential time for secondary steam releases to atmosphere, carrying with the steam a portion of the activity in the steam generators.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary-to-secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS, and contributes to increased post-accident doses. If secondary specific activity cannot be restored to within limits 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 operating 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 ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)	analysis assumptions as to the source terms in post-accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31-day Frequency 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 non-essential 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 sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection actuation 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, are presented in Reference 1. The principal safety-related function of the CCW System is the removal of decay heat from the reactor via the [Shutdown Cooling System (SCS) heat exchanger]. This may utilize the SCS heat exchanger, during a normal or post-accident cooldown and shutdown, or the Containment Spray System during the recirculation phase following a loss-of-coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one CCW train in conjunction with a 100%-capacity Containment Cooling System (containment spray, containment coolers, or a combination) removing core decay heat 20 minutes after a design basis LOCA. This prevents the containment sump fluid from increasing in temperature during the recirculation phase

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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 safety injection 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 [SCS] 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 [SCS] trains operating. One train is sufficient to remove decay heat during subsequent operations with T_{cold} < [200]°F. This assumes that a maximum seawater temperature of [76]°F occurs 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 single active failure occurs coincident with the loss-of-offsite power.

A train is considered OPERABLE when:

a. Its pump and associated surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger and 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 systems inoperable, but does not affect the OPERABILITY of the CCW

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BASES (continued)

LCO (continued)

System. [For this facility, the following support systems are required to be operable to ensure CCW System OPERABILITY:]

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

[For this facility, the main systems supported by the CCW System and the justification for not declaring the main systems inoperable upon failure of the CCW System are as follows:]

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system that must be prepared to perform its post-accident safety functions, primarily RCS heat removal by cooling the [SCS] heat exchanger.

In MODES 5 and 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

With one CCW train inoperable, 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.

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

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BASES (continued)

ACTIONS (continued)

that needed to be declared inoperable upon the failure of one or more support featur;s 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 supported systems' LCOs 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 c CCW System loss-of-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 if 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, action must be taken immediately to restore at least one train to OPERABLE status. In this case, there is no heat sink for the [SCS], 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.

With both trains inoperable, flexibility is left to the operator (and abnormal operating procedures) 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 restored to OPERABLE

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BASES (continued)

ACTIONS (continued)

status, then the plant should be placed in MODE 5. In this case, LCO 3.0.3 is not applicable as 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 potentially being mispositioned are in their correct position.

The 31-day Frequency of this SR 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 control governing valve operation, and as a means of providing added assurance of correct valve positions.

SR 3.7.7.2

This SR 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 since it was considered prudent that many surveillances be performed only during a plant outage. This was due to the plant conditions needed to perform the SR, and the potential for unnecessary plant transients if the SR is performed with

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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.

SR 3.7.7.3

This SR 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 since it is considered prudent that many surveillances be performed only during a plant outage. This was 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.

REFERENCES

1. [Unit Name] FSAR, Section [9.2.2], "[Component Cooling Water System]."

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 or a normal shutdown, the SWS also provides this function for various safety-related and non-safety-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, valves, instrumentation, and two cyclone separators. The pumps and valves are remote 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 actuation signal (SIAS) 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 (AFW) 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 (containment spray, containment coolers, or a combination), removing core decay heat 20 minutes following a design basis LOCA (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA

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APPLICABLE SAFETY ANALYSES (continued)

and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the safety injection pumps. The SWS is designed to perform its function with a single failure of any active component, assuming the loss-of-offsite power.

The SWS, in conjunction with the CCW System, also cools the plant from [shutdown cooling] (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 [Shutdown Cooling System] trains that are operating. One SWS train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes that 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 the worst single active failure occurs coincident with the loss-of-offsite power.

A train is considered OPERABLE when:

a. It has an OPERABLE pump; and

b. The associated piping, valves, instrumentation, heat exchanger, and cyclone separator on the safety-related flow path are OPERABLE.

The isolation of the SWS from other components or systems not required for safety, may render those components or systems inoperable, but does not affect the OPERABILITY of the SWS. [For this facility, the main systems supported by SWS and the justification for not declaring the main systems inoperable upon failure or SWS are as follows:]

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

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BASES (continued)

LCO (continued)

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

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS System is a normally operating system, which must be prepared to perform its post-accident safety functions, primarily RCS heat removal, by cooling the CCW System, and thus the [Shutdown Cooling System].

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

ACTIONS

A.1

If one SSW 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 taking into account 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.

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

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ACTIONS (continued)

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 SWS train inoperable, and one or more required support or supported features inoperable associated with the other redundant SWS train, enter 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 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 [Shutdown Cooling System]; thus, restore one SWS train 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. 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 locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31-day Frequency of this SR 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 control governing valve operation, and to provide added assurance of correct valve positions.

SR 3.7.8.2

This SR demonstrates proper automatic operation of the SWS valves. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing. 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 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 3.7.8.3

The SR demonstrates proper automatic operation of the SWS pumps. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. 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

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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.

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-safety-related components during a transient or accident. This is done utilizing the Service Water System (SWS).

The UHS has been defined as that complex of water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting tne 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 another source(s) and a make-up system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a one-day supply, the systems for replenishing the basin and the back-up source(s) become of sufficient importance that the make-up system itself may be required to meet the same design criteria as an engineered safety feature (ESF) (e.g., single-failure considerations, and multiple make-up water sources may be required).

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BASES (continued)

BACKGROUND (continued)

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 make-up requirement. [Development of UHS technical specifications for plants with cooling towers and makeup systems may require Actions and SRs for components in addition to cooling tower fans (e.g., make-up 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 operations occurrences (AOOs) in which the plant is cooled down and placed on [shutdown cooling]. [For those plants using 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 are required to remove the core decay heat.

The operating limits are based on conservative heat transfer analyses for the worst-case LOCA. Reference 1 provides the details of the assumptions used in the analysis. The

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

assumptions include: worst-expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst-case 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 to be OPERABLE to ensure UHS OPERABILITY:]

[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:]

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

APPLICABILITY

In MODES 1, 2, 3, and 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.

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BASES (continued)

APPLICABILITY (continued)

In MODES 5 and 6, the OPERABILITY requirements of the UHS are determined by the systems they support.

ACTIONS

A.1 and A.2

Required Action A.1 assures the required cooling capacity will be available should a Design Basis Accident (DBA) occur.

This action may be satisfied by examining logs or other information to determine if the cooling tower 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, with one cooling tower fan per cooling tower inoperable, the inoperable cooling tower fans must be restored to OPERABLE status within 7 days, or action must be taken to reduce power. The spe fied Completion Time is consistent with other LCOs for the loss of one-half of a 200%-capacity train of an Engineered Safety Features (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 cooling tower fans within a Completion Time of [ ] hours.

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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 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 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.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR ensures adequate long-term (30 days) cooling can be maintained. The level specified ensures enough NPSH available for operating the SWS Pumps. The 24-hour

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES.

SR 3.7.9.2

This SR verifies that the SWS can cool the CCW System to at least its maximum design temperature within the maximum accident or normal design heat loads for 30 days following a DBA. The [24]-hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable Modes.

SR 3.7.9.3

Operating each cooling tower fan for $\geq$ 15 minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration can be detected for corrective action. The 31-day Frequency was developed considering 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, and 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 to a 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 Specification, is that there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel-handling accident. With 23 ft, the assumptions of 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 above the top of the bundle and the surface, by the width of the bundle. To offset this small non-conservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first [few] rows fail from a hypothetical maximum drop. The fuel storage pool water level satisfies Criterion 3 of the NRC Interim Policy Statement.

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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 water level constitutes the following:]

[For this facility, the following support systems are required to be OPERABLE to ensure fuel storage pool water level OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not require declaring 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 an accide cannot be met, steps should be taken to preclude the accic t from occurring. With the fuel storage pool leve ower than the required level, the movement of spent fuel immediately brought to a halt in a safe position. This effectively precludes a spent-fuel-handling accident from occurring. In such a case, plant procedures control the movement of loads over the spent fuel.

In the event that the required fuel storage pool water level channels are found inoperable, the fuel storage pool water level is considered to be not within limits, Required Action A.1 and Required Action A.2 apply.

A.2

Action to restore the water level should commence immediately, and be carried through to completion.

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BASES (continued)

ACTIONS (continued)

Required Action A.1 and Required Action A.2 are modified by a Note which allows an exemption to LCO 3.0.3 and LCO 3.0.4. Both LCOs are not applicable, as events in the fuel storage pool are not affected by MODE level or facility operations.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

This SR verifies 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 at equilibrium with that of 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."

5. Title 10, Code of Federal Regulations, Part 100.11, "Determinations 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 safety grade method for cooling the plant to Shutdown Cooling System (SCS) 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 Bypass System.

Four ADV lines are provided. Each ADV line consists of one ADV and an associated block valve. Two ADV lines per steam generator are required to meet single-failure assumptions following an event rendering one steam generator unavailable for Reactor Coolant System (RCS) heat removal.

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 which, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ADVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the ADVs for the time required for RCS cooldown to SCS 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, hand wheels 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 SCS entry conditions. The

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

design rate of [75]°F/hour is applicable for both steam generators, each with two ADVs. This design is adequate to cool the plant to SCS entry conditions with only one ADV and one steam generator, 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 SCS entry conditions for accidents accompanied by a loss-of-offsite power. Prior to the operator action, the main steam safety valves (MSSVs) are used to maintain the steam generator's pressure and temperature at the MSSVs setpoint. This is typically 30 minutes following the initiation of an event. (This may be less for a steam generator tube rupture (SGTR) event.) The limiting events are those that render one steam generator unavailable for RCS heat removal, with a coincident loss-of-offsite power; this results from a turbine trip and the single failure of one ADV on the unaffected steam generator. Typical initiating events falling into this category are a main steam line break (MSLB) upstream of the main steam isolation valves, a feedwater line break (FWLB), and an SGTR event (although the ADVs on the affected steam generator may still be available following a SGTR event).

For the recovery from an SGTR event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary-to-secondary break flow into the ruptured steam generator. After the break flow is terminated, the operator would then continue the cooldown to SCS conditions, similar to the cooldown for other events. The operator is assumed to use only the ADVs on the non-ruptured steam generator to perform the limited cooldown required to terminate the break flow, and subsequently to cool down the plant to SCS entry conditions. The time required to terminate the primary-to-secondary break flow for an SGTR is more critical than the time required to cool down to SCS 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 consideration of any single-failure assumptions regarding the failure of one ADV to open on demand.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The design must accommodate the single failure of one ADV to open on demand; thus, each steam generator must have at least two ADVs. 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

Two ADV lines are required on each steam generator to ensure that at least one ADV is OPERABLE to conduct a plant cooldown following an event in which one steam generator becomes unavailable, accompanied by a single-active failure of one ADV line on the unaffected steam generator. The block valves must be OPERABLE to isolate a failed open ADV. 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 SCS 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 a controlled relief of the main steam flow, and is capable of fully opening and closing on demand.

[For this facility, an OPERABLE ADV line constitutes the following:]

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

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

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the ADV lines provide the path for cooling the RCS to SCS entry conditions following an SGTR.

In MODES 5 and 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, and a non-safety-grade backup in the Steam Bypass System and 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. As 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 and 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR 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 SR when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.7.11.2

This SR verifies the OPERABILITY of the block valves. The function of the block valve is to isolate a failed open ADV. Cycling the block valve 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 SR 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 Air Cleanup System (CREACS)

BASES

BACKGROUND

The CREACS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The CREACS 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 iodine), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as do demisters that remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines, and to back up the main HEPA filter bank if it fails.

The CREACS is an emergency system, part 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 CREACS places the system into either of two separate states 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 control room air through the redundant trains of HEPA and charcoal filters. The emergency radiation state initiates pressurization and filtered ventilation of the air supply to the control room.

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BASES (continued)

BACKGROUND (continued)

Outside air is filtered, [diluted with building air from the electrical equipment and cable-spreading rooms,] and then 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 MODE, preventing 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 provides an air exchange rate in excess of [25]% per hour. The CREACS 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 one damper's failure to shut will not result in a breach of isolation. The CREACS is designed in accordance with Seismic Category I requirements.

The CREACS is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5-rem whole-body dose.

APPLICABLE SAFETY ANALYSES

The CREACS components are arranged in redundant safety-related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

During emergency operation, the CREACS maintains the temperature between [70]°F and [85]°F. The CREACS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss-of-coolant accident fission-product release presented in 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 CREACS, assuming a loss-of-offsite power, does not impair the ability of the system to perform its design function.

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

LCO

Two independent and redundant trains of the CREACS are required to ensure that at least one is available, assuming that a single failure disables the other train. Total system failure could result in a control room operator receiving a dose in excess of 5 rem in the event of a large radioactive release.

The CREACS is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both trains. A train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filters 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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(continued)

BASES (continued)

LCO (continued)

In addition, the control room boundary must be maintained including the integrity of the walls, floors, ceilings, ductwork, and access doors.

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

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

APPLICABILITY

In MODES 1, 2, 3 and 4, the CREACS must be OPERABLE to control operator exposure during and following a DBA.

[In MODES 5 and 6, the CREACS may be required to cope with the release from a rupture of an outside waste-gas tank.]

During movement of irradiated fuel, the CREACS must be OPERABLE to cope with the release from a fuel-handling accident.

ACTIONS

A.1

With one CREACS train inoperable, the inoperable CREACS train must be restored to OPERABLE status within 7 days. During this period, the remaining OPERABLE CREACS subsystem is adequate to perform control room radiation protection function. The 7-day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

Concurrent failure of two CREACS trains would result in the loss-of-function capability. Therefore, LCO 3.0.3 must be immediately entered.

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BASES (continued)

ACTIONS (continued)

B.1 and B.2

In MODE 1, 2, 3, or 4, if Required Action A.1 cannot be completed within the required Completion Time, the plant must be placed in a MODE that minimizes the accident 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, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CREACS train should be immediately placed in the emergency mode of operation [toxic-gas isolation state]. This action ensures that the remaining train is OPERABLE, no failures that would prevent automatic actuation will occur, and any active failure will be readily detected.

Required Action C.1 is modified by a Note to place the system in the emergency mode if the 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 the accident risk, but does not preclude the movement of fuel to a safe position.

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

When in MODES 5 and 6, and during movement of irradiated fuel with two CREACS 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 the accident risk, but does not preclude the movement of fuel to a safe position.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR 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 that they start and function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operations dry 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 3.7.12.2

Specification 5.8.4.5 of the Ventilation Filter Testing Program (VFTP) encompasses all the CREACS filter tests consistent with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing HEPA filter performance, charcoal absorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR demonstrates that each CREACS train starts and operates on an actual or simulated actuation signal. The Frequency of 18 months is specified in Regulatory Guide 1.52 (Ref. 3).

SR 3.7.12.4

This SR demonstrates the integrity of the control room enclosure and the assumed inleaking rates of potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the CREACS. During the emergency radiation state of the emergency mode of operation, the CREACS is designed to pressurize the control room to [0.125] inches water gauge positive pressure

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

with respect to adjacent areas in order to prevent unfiltered inleakage. The CREACS 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.

SR 3.7.12.5

[For this facility, the purpose of this SR is as follows:]

REFERENCES

1. [Unit Name] FSAR, Section [9.4], "[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 Air Cleanup 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 to maintain the control room temperature is discussed in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the CREHVAC is to maintain hability of the control room environment throughout 30 days of 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, assuming 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, considering equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

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

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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 disables 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 that are necessary to maintain the control room temperature are OPERABLE in both trains. These components include the 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 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 MODES 1, 2, 3, 4, and during movement of irradiated fuel, the CREHVAC must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY requirements following isolation of the control room.

In MODES 5 and 6, CREHVAC may not be required for those facilities which do not require automatic control room isolation.

ACTIONS

A.1

With one CREHVAC train 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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BASES (continued)

ACTIONS (continued)

required capabilities, and the alternate safety- or non-safety-related cooling means that are available. [For this facility, the alternate cooling means are as follows:]

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 accident 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 any active failure will 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 could result in a release of radioactivity that might require isolation of the control room. This places the plant in a condition that minimizes the accident risk, but it does not preclude the movement of fuel to a safe position.

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

In MODE 5 or 6, or during movement of irradiated fuel, with two CREHVAC trains inoperable, the Required Action is to immediately suspend activities that could result in a

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BASES (continued)

ACTIONS (continued)	release of radioactivity that might require isolation of the control room. This places the plant in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.
SURVEILLANCE REQUIREMENTS	SR 3.7.13.1 This SR verifies that the heat-removal capability of the system is sufficient to meet design requirements. This SR is performed at a Frequency of 18 months and consists of a combination of testing and calculations. An 18-month Frequency is appropriate because significant degradation of the CREHVAC is slow and is not expected over this time period.
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 and lower region of the auxiliary building following receipt of a Safety Injection Actuation Signal or Coolant Injection Actuation Signal.

The ECCS PREACS is a standby system, parts of which may also operate during normal plant operations. The Reactor Auxiliary Building Main Ventilation System provides normal cooling. During emergency operations, the ECCS PREACS dampers are realigned and fans are started to initiate filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the ECCS pump room, the pump room 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.

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BASES (continued)

BACKGROUND (continued)

The ECCS PREACS is discussed in several sections of the FSAR (Refs. 1, 2, and 3) as it may be used for normal, as well as post-accident, atmospheric cleanup functions. The primary purpose of the heaters is to maintain the relative humidity to 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 safety injection pump-seal failure, during the recirculation mode. In such a case, the system limits the radioactive release to within 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 a single failure disables the other train coincident with a loss-of-offsite power. Total system failure could result in the atmospheric release from the ECCS pump room exceeding the 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, 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:]

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

APPLICABILITY

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

In MODES 5 and 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.

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BASES (continued)

ACTIONS (continued)

The 7-day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72-hour Completion Time) and this system is not a direct support system for the ECCS. The 7-day Completion Time is based on the low probability of a DBA occurring during this time period, and the considerat that the remaining train can provide the required capa. y.

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 Pump Room Exhaust Air Cleanup System 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 ECCS PREACS train inoperable, and one or more required support or supported features inoperable associated with the other redundant ECCS PREACS; 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

This SR 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. As 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 filer 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 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

This SR 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 function 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 Section 6.5.1 of NUREG-0800 (Ref. 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 is not excessive.

The number and depth of the adsorber elements ensures 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 magnitude of the pressure drop indicates acceptable performance and is based on manufacturer's recommendations for the filter and adsorber elements at the design flow rate. An increase in pressure

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

drop, or 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. A 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. 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. 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 Systems," 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 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 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, part of which may also be operated during normal plant operations. Upon receipt of the actuating signal, normal air discharges from the fuel-handling 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.

(continued)

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 absorbed 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 remaining one 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. 5).

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 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 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 functions;

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained; and

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BASES (continued)

LCO (continued)

d. SRs are met.

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

[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 MODES 1, 2, 3, and 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 MODES 5 and 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 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 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 in MODE 5 within 36 hours. The allowed Completion Times are

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(continued)

BASES (continued)

ACTIONS (continued)

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 dur. 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

This SR 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. 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

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(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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. 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 is discussed in detail in the VFTP.

SR 3.7.15.3

This SR 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. 4).

SR 3.7.15.4

This SR 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 [3,000] cfm to the fuel building. The Frequency of 18 months is consistent with the guidance provided in Section 6.5.1 of NUREG-0800 (Ref. 6).

The minimum system flow rate maintains a negative pressure in the fuel building, and provides sufficient air velocity to transport particulate contaminants, assuming only one filter train is in operation.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 is not excessive.

The number and depth of the adsorber elements ensures 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 manufacturer's recommendations for the filter and adsorber elements at the design flow rate. An increase in pressure drop, or decrease in flow indicates that the filter is being loaded or 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 3.7.15.5

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. 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 [15.7.4], "[Fuel-handling Accident]."

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BASES (continued)

REFERENCES (continued)

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. 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."

6. NUREG-0800, Section 6.5.1, "Standard Review Plan," Rev. 2, "ESF Atmosphere Cleanup System," July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.16 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 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, which follows the adsorber section, collects carbon fines and provides backup in case of failure 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 Actuation System or Coolant Injection Activation System 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 Engineered Safety Feature Actuation System 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 or demisters 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) as 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 a valve packing leakage during a Design Basis Accident (DBA). In such a case, the system restricts the radioactive release to within 10 CFR 100 (Ref. 5) 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 or an excessive LEAKAGE. Either type of failure may result in less efficient removal for any gaseous or particulate material released to the penetration rooms 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 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 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 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 MODES 1, 2, 3, and 4, the PREACS is required to be OPERABLE, consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).

In MODES 5 and 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 for the ECCS (72 hours 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 consideration 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 a normal manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR 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 dries 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

Specification 5.8.4.p of 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 the performance of the HEPA filter, 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 demonstrates that each PREACS trains 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 demonstrates the integrity of the penetration room enclosure. The ability of the penetration room to maintain negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of the PREACS. During the emergency mode of operation, PREACS is designed to maintain a slightly negative pressure at a flow rate of [3000] 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 Section 6.5.1 of NUREG-0800 (Ref. 6).

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 is operating.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 is not excessive.

The number and depth of the absorber elements ensures 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 manufacturer's recommendations for the filter and absorber elements at the design flow rate. An increase in pressure drop, decrease in flow indicates that the filter is being loaded or is indicative of other problems with the system.

This test is conducted with the tests for filter penetration, and thus an 18-month Frequency, consistent with that specified in Regulatory Guide 1.52 (Ref. 4), is used.

SR 3.7.16.5

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]."

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BASES (continued)

REFERENCES (continued)

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," U.S. Nuclear Regulatory Commission.

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 Systems," July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.17 Essential Chilled Water (ECW)

BASES

BACKGROUND	The ECW System provides a heat sink for the removal of process and operating heat from selected safety-related air-handling systems during a transient or accident. The ECW System is a closed-loop system consisting of two independent trains. Each 100%-capacity train includes a heat exchanger, surge tank, pump, chemical addition tank, piping, valves, controls, and instrumentation. An independent 100%-capacity chilled water refrigeration unit cools each train. The ECW System is actuated on a safety injection actuation signal (SIAS) and supplies chilled water to the heating, ventilation, and air conditioning (HVAC) units in EMERGENCY SAFETY FEATURE (ESF) equipment areas (e.g., the main control room, electrical equipment room, and safety injection pump area). The flow path for the ECW System includes the closed loop of piping to all serviced equipment, and branch lines up to the first normally-closed isolation valve. During normal operation, the normal HVAC System performs the cooling function of the ECW System. The normal HVAC System is a non-safety-grade system that automatically shuts down when the ECW System receives a start signal. Additional information about the design and operation of the system, along with a list of components served, can be found in Reference 1.
APPLICABLE SAFETY ANALYSES	The design basis of the ECW System is to remove the post-accident heat load from ESF spaces following a Design Basis Accident coincident with a loss-of-offsite power. Each train provides chilled water to the HVAC units at the design temperature of 42°F and flow rate of 400 gpm. The maximum heat load in the ESF pump room area occurs during the recirculation phase following a loss-of-coolant accident (LOCA). During recirculation, hot fluid from the

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

containment pump is supplied to the high-pressure safety injection and containment spray pumps. This heat load to the area atmosphere must be removed by the ECW System to ensure that these pumps remain OPERABLE.

The ECW System satisfies Criterion 3 of the NRC Interim Policy Statement.

LCO

The requirements for two ECW trains provide the required redundancy to ensure that the system functions to remove post-accident heat loads, assuming the worst-single failure.

A train is considered OPERABLE when:

a. Its pump and associated surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger, refrigeration unit, and instrumentation on the safety-related flow path are OPERABLE.

The isolation of the ECW from other components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the ECW System. [For this facility, these components or systems are as follows:]

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

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

[For this facility, the supported systems impacted by the inoperability of the ECW system and the justification of whether or not each supported system is declared inoperable are as follows:]

APPLICABILITY

In MODES 1, 2, 3, and 4, the ECW System is required to be OPERABLE when a LOCA or other accident would require ESF operation.

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BASES (continued)

APPLICABILITY (continued)

In MODES 5 and 6, potential heat loads are smaller and the probability of accidents requiring the ECW System is low.

ACTIONS

A.1

If one ECW train is inoperable, it must be restored to OPERABLE status within 7 days. In this condition, the [one] OPERABLE ECW train is adequate to perform the cooling function. The 7-day Completion Time is appropriate because of the low probability of an event during this time, the 100% capacity OPERABLE ECW train, and the redundant availability of the normal HVAC System.

B.1

With one ECW train inoperable, verify that the Required Actions have been initiated for those supported systems declared inoperable by the support ECW 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 ECW train 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 ECW train inoperable, and one or more required support or supported features inoperable associated with the other redundant ECW trains, enter Required Actions of Condition D. Condition C is indicative of a loss of ECW System functional capability.

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BASES (continued)

ACTIONS (continued)

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

If the ECW train cannot be restored to OPERABLE status within the associated Completion Time, or two ECW 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 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 3.7.17.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the ECW flow path provides assurance that the proper flow paths exist for ECW 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 locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. The 31-day Frequency of this SR 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 control governing valve operation, and the added assurance of valve correct positions.

SR 3.7.17.2

This SR demonstrates proper automatic operation of the ECW System components. This surveillance ensures that the ECW pumps will start in the event of any accident or transient that generates a SIAS actuation signal. This SR also ensures that each automatic valve in the flow paths actuate to its correct position on an actual or simulated SIAS signal. The ECW System cannot be fully actuated as part of the SIAS CHANNEL FUNCTIONAL TEST during normal operation. The actuation logic is tested as part of the SIAS functional

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

test every 92 days, except for the subgroup relays that actuate the system that cannot be tested during normal plant operation. The surveillance interval of 18 months is based on the plant conditions needed to perform the surveillances, and the potential for unplanned transients if the surveillance is performed at power. The 18-month Frequency is also acceptable based on the design reliability and confirming operating experience of the equipment.

REFERENCES

1. [Unit Name] FSAR, Section [9.2.9], "[Essential Chilled Water System]."

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)

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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-P[illegible] [PLANT SPECIFIC: In general, {Division 3] does no[illegible] a load sequencer since it has only one large load, i.[illegible]igh 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 operability 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 operability above, they are, in fact, not operable). Instead, the definition of operability 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 operability 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 $\leq$ BX $\leq$ 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 $\geq$ 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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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 Electric 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)

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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)

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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 assumptions in the 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.

SR 3.8.1.6

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 ≥ 0.83° but ≤ 0.89° or an API gravity at 60°F of ≥ 27° but ≤ 39°, a kinematic viscosity at 40°C of ≥ 1.9 centistokes but ≤ 4.1 centistokes, and a flash point ≥ 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 during or following DG testing. In such a case a 31-day Frequency is appropriate. Since proper operation of fuel-transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs. 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.

SR 3.8.1.20

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 proper 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 Surveillance and is intended to be consistent with expected fuel-cycle lengths.

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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] $\leq$ 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.

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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 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 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 is 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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(continued)

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]. (continued)

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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:]

<u>B.1</u>

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.

<u>C.1</u>

With one [division] DC electrical power subsystem inoperable <u>AND</u> 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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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 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).

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(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:]

(continued)

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 1[illegible] "Accident Analyses"], assume ESF systems are OPERABL[illegible] he 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 tha upport.} 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. (continued)

(continued)

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 [illegible] 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 fol'ows:]

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

<u>A.1, A.2, A.3, and A.4</u>

With ... required inverter inoperable, its associated AC vital bus ... es 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: <u>OR</u> [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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(continued)

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, ar the items listed above should be considered in speci ng this time for a given facility]}. The 24-hour limit i based upon engineering judgment, ta ing 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

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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

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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 the [Division 1, 2, and 3] AC and DC electrical power distribution 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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(continued)

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 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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(continued)

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 System

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 [illegible]-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 [NK03] from battery [NN13] and charger [NK23] Distribution Panels [NK41, NK43, NK51]	Bus [NK02] from battery [NK12] and charger [NK22] Bus [NK04] from battery [NK14] and charger [NK24] Distribution Panels [NK42, NK44, NK52]	Bus [NK05] from battery [NK15] and charger [NK25] Distribution Panel [NK45]
AC vital buses	[120 V]	Bus [NN01] from inverter [NN11] connected to bus [NK01] Bus [NN03] from inverter [NN13] connected to bus [NK03]	Bus [NN02] from inverter [NN12] connected to bus [NK02] Bus [NN04] from inverter [NN14] connected to bus [NK04]	Bus [NN05] from inverter [NN15] connected to bus [NK05]

* 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 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, [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 designeo 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, tnis 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 arc 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 under all conditions. 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 counters 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 $K_{eff} \leq 0.95$ during fuel handling, with control element assemblies (CEAs) 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 tank (RWT) into the open reactor vessel by gravity feeding or by the use of the Shutdown Cooling (SDC) System pumps.

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BASES (continued)

BACKGROUND (continued)

During refueling, the water volumes in the RCS, the refueling cavity, and the refueling canal are contiguous. 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 SDC System in the RCS, and the natural circulation due to thermal driving heads in the reactor vessel and the refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The SDC System is in operation during refueling (see LCO 3.9.4 and LCO 3.9.5) to provide forced circulation in the RCS and assist to 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 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.

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BASES (continued)

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 a core K_{eff} of ≤ 0.95 is maintained during fuel-handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $K_{eff} \leq 0.95$. 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 to maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the 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 the movement of a component to a safe condition.

A.3

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, there is no unique design basis event that must be satisfied. The only requirement is to

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BASES (continued)

ACTIONS (continued)

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.

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 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 Nuclear Instrumentation

BASES

BACKGROUND

The source range monitors are used during refueling operations to monitor the core reactivity condition. The installed source range 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 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 five decades of neutron flux (1E+5 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 monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The corrective action for such an event is for the operator to close the primary water makeup valve that provides makeup water to the Reactor Coolant System (RCS). The analysis of the uncontrolled boron dilution accident shows that the time for operator corrective action to terminate the event is greater than the 30 minutes required in Reference 3.

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

(continued)

BASES (continued)

LCO

This LCO requires two source range monitors are OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. The OPERABILITY of the monitors is established via a CHANNEL CHECK and CHANNEL FUNCTIONAL 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 monitor OPERABILITY:]

[For this facility, those required support systems which, upon their failure, do not declare the source range monitor inoperable and their justification are as follows:]

APPLICABILITY

In MODE 6, the source range monitors must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels.

[Digital—In MODES 2, 3, 4, and 5, with the Reactor Trip Circuit Breakers closed, the comparably installed source range detectors and circuitry that are used are the Logarithmic Power Level—High. These are required to be OPERABLE by LCO 3.3.1, "Reactor Protective System Instrumentation." In MODES 3, 4, and 5 with the reactor trip circuit breakers open, the [logarithmic] power monitoring instruments are required to be OPERABLE by LCO 3.3.6.]

[Analog—In MODES 2, 3, 4, and 5, with the reactor trip circuit breakers closed, the comparably installed source range detectors and circuitry that are used are the Power Rate of Change—High. These are required to be OPERABLE by LCO 3.3.1, "Reactor Protective System Instrumentation." In MODES 3, 4, and 5 with the reactor trip circuit breakers open, the wide range power instruments are required to be OPERABLE by LCO 3.3.6.]

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With only one source range 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 monitor OPERABLE, action shall be initiated to restore the inoperable monitor to OPERABLE status within 7 days. Seven days is a reasonable period of 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 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 monitor is restored to OPERABLE status. The 15-minute Completion Time is allowed for an operator to initiate corrective actions.

B.2

With no source range 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 monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to verify that the required boron concentration exists.

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BASES (continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 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 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 Protection System."

SR 3.9.2.2

The performance of a CHANNEL FUNCTIONAL TEST provides assurance that the analog process control equipment and trip setpoints are within limits. [For this Facility, CHANNEL FUNCTIONAL 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.

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BASES (continued)

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]."

3. 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 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS movement of fuel assemblies within containment with irradiated fuel in containment, a release of fission-product radioactivity within the 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 structure 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 four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1 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

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BASES (continued)

BACKGROUND (continued)

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

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

The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a [42]-inch purge penetration and a [42]-inch exhaust penetration. The second subsystem, 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 exchanges are necessary to conduct refueling operations. The normal [42]-inch purge system is used for this purpose and all 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 which provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by and OPERABLE automatic isolation valve, by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements.
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 3.9.6, "Refueling Water Level," and the minimum decay time of [72] hours prior to CORE ALTERATIONS ensure that the release of fission-product radioactivity, subsequent to a fuel-handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan Section 15.7.4, Rev. 1 (Ref. 1) 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

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BASES (continued)

LCO (continued)

from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the 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 assure releases through the valves are terminated such that the 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 penetration 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

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the Containment Purge and Exhaust

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BASES (continued)

ACTIONS (continued)

Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the plant must 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 3.9.3.1

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 that will ensure 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 the 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 3.9.3.2

This surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18-month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.4 [Digital or 3.3.3 Analog], "Miscellaneous Actuations," the Containment Purge Isolation Signal System requires a CHANNEL CHECK every

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

7 days and a CHANNEL FUNCTIONAL TEST every 31 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice 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 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation—High Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS) as required by GDC 34, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s) where the heat is transferred to the Component Cooling Water (CCW) System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC 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 SDC 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 SDC 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 a 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 would eventually challenge the integrity of the fuel cladding, which is a fission-product barrier. One train of the SDC System is required to be operational in MODE 6 with the water level ≥ 23 ft above the top of the reactor vessel flange to prevent this challenge. The LCO does permit de-energizing of the SDC pump for short durations under the condition that the boron concentration is not diluted. To ensure that the coolant temperature

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

remains < 200°F, short durations of pump de-energization would be repeated only a few times. This conditional de-energizing of the SDC pump does not result in a challenge to the fission-product barrier.

Although the SDC 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 SDC System is retained as a technical specification.

LCO

Only one SDC loop is required for decay-heat removal in MODE 6 with a water level ≥ 23 ft above the top of the reactor vessel flange. Only one SDC 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 SDC loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC loop includes an SDC 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 SDC loop in operation constitutes the following:]

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

[For this facility, those required support systems which, upon their failure, do not declare the SDC 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 SDC System and their justification of whether or not each supported system is declared inoperable are as follows:] It should be noted that LCO 3.9.4 may need to be augmented with additional Conditions if it is determined that the SDC System provides support to other systems in technical specifications during this MODE of operation.

The LCO is modified by a Note that allows the required operating SDC loop to be removed from service for up to one hour in each 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-SDC 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 SDC loop must be OPERABLE and in operation in MODE 6 with the water level $\geq$ 23 ft above the top of the reactor vessel flange to provide decay-heat removal. The 23-foot level was selected because it corresponds to the 23-foot requirement established for fuel movement in LCO 3.9.6, "Refueling Water Level." Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, "Reactor Coolant System," and Section 3.5, "Emergency Core Cooling System." SDC loop requirements in MODE 6, when the water level is < 23 ft above the top of the reactor vessel flange, are located in LCO 3.9.5, "Shutdown Cooling and Coolant Circulation—Low Water Level."

ACTIONS

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

A.1

If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur through the addition of water with a lower boron

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BASES (continued)

ACTIONS (continued)

concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

A.2

If SDC loop requirements are not met, actions shall be tak immediately to suspend operations involving an increase i reactor decay-heat load. With no forced circulation cooling, decay-heat removal from the core occurs by natura 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 the decay-heat load, such as loadi g a fuel assembly, is a prudent action under this condition.

A.3

If SDC loop requirements are not met, actions shall be taken and continued in order to satisfy SDC 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 operator to initiate corrective actions.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This surveillance verifies that the SDC 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 SDC System.

REFERENCES

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

B 3.9 REFUELING OPERATIONS

B 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS) as required by GDC 34, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s) where the heat is transferred to the Component Cooling Water (CCW) System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC 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 SDC 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 SDC 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 would eventually challenge the integrity of the fuel cladding, which is a fission-product barrier. Two trains of the SDC System are required to be OPERABLE and one train is required to be in operation to prevent this challenge.

Although the SDC 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

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	reduction. Therefore, the SDC System is retained as a technical specification.

LCO

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

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC loop consists of an SDC 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 SDC loop in operation constitutes the following:]

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

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

[For this facility, the supported systems impacted by the inoperability of an SDC 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 SDC System provides support to other systems included in technical specifications during this MODE of operation.

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BASES (continued)

APPLICABILITY

Two SDC loops are required to be OPERABLE, and one SDC 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 SDC System in other MODES are covered by LCOs in Section 3.4, "Reactor Coolant System." MODE 6 requirements with a water level ≥ 23 ft above the reactor vessel flange are covered in LCO 3.9.4, "Shutdown Cooling and Coolant Circulation—High Water Level."

ACTIONS

A.1 and A.2

If one SDC loop is inoperable or not in operation, actions shall be initiated and continued until the SDC 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 established at ≥ 23 ft above the reactor vessel flange, the Applicability will change to that of LCO 3.9.4 "Shutdown Cooling and Coolant Circulation—High Water Level," and only one SDC 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 SDC loop is OPERABLE or in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

B.2

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

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BASES (continued)

ACTIONS (continued)

If no SDC loop is OPERABLE or in operation, alternate actions shall have been initiated within 15 minutes under Condition A 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 decay heat removal using the charging or safety injection pumps through the Chemical and Volume Control System (CVCS) with consideration for the boron concentration. The method used to remove decay heat should be the most prudent as well as the safest choice, based 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 SDC 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 events also should 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 3.9.5.1

This surveillance verifies that one SDC loop is OPERABLE, in operation, and circulating reactor coolant. The flow rate

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

is determined by the flow rate necessary to provide sufficient decay-heat removal capability and to prevent thermal and boron stratification in the core. In addition, this surveillance verifies that the other SDC loop is OPERABLE.

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

REFERENCES

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

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling 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, the refueling cavity, the refueling canal, the fuel-transfer canal, and the spent-fuel pool. Sufficient water is necessary to retain iodine fission-product activity in the water in the event of a fuel-handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident 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 refueling canal is an initial condition design parameter in the analysis of the fuel-handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel-rod iodine inventory (Ref. 1).

The fuel-handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft, and a minimum decay time of 72 hours prior to fuel-handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel-handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).

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

(continued)

BASES (continued)

LCO

A minimum refueling 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 water level OPERABILITY:]

[For this facility, those required support systems which upon their failure do not declare the refueling water level inoperable and their justification are as follows:]

APPLICABILITY

Within the containment, LCO 3.9.6, "Refueling 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 3.7.10, "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 water level indication channels are found inoperable, the refueling water level is considered to be not within limits and Required Action A.1 applies.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the postulated fuel-handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of a 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, Section 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)

APPENDIX A (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)

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)

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

(continued)

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 TEST 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)

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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
SDC	shutdown cooling
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

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1432
Vol. 3

2. TITLE AND SUBTITLE

Standard Technical Specifications
Combustion Engineering Plants

Bases (Sections 3.4-3.9)

Draft Report for Comment

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	1991

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

DRAFT

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Operational Events Assessment
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This draft report documents the results of the NRC staff review of new Standard Technical Specifications (STS) proposed by the Combustion Engineering 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.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Technical Specifications
Combustion Engineering
PWR

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)